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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

December 7, 2005

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

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ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

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528^{TH} MEETING

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WEDNESDAY,

DECEMBER 7, 2005

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The Committee met in Room T-2B3 of the U.S. Nuclear Regulatory Commission, Two White Flint North, 11545 Rockville Pike, Rockville, Maryland, at 1:00 p.m., Graham B. Wallis, Chairman, presiding.

<u>COMMITTEE MEMBER PRESENT</u>:

GRAHAM B. WALLIS, ACRS Chairman
WILLIAM J. SHACK, ACRS Vice Chairman
JOHN E. SIEBER, ACRS Member-at-Large
MARIO V. BONACA, ACRS Member
RICHARD S. DENNING, ACRS Member
THOMAS S. KRESS, ACRS Member
DANA A. POWERS, ACRS Member
VICTOR H. RANSOM, ACRS Member

C O N T E N T S

P R O C E E D I N G S

(1:03 p.m.)

CHAIRMAN WALLIS: Good afternoon. The meeting will now come to order.

This is the first day of the 528th meeting of the Advisory Committee on Reactor Safeguards. During today's meeting the committee will consider the following: final review of Vermont Yankee extended power uprate application and the associated safety evaluation; draft ACRS report on the NRC Safety Research Program; and preparation for meeting with the NRC Commissioners, which will be tomorrow, the actual meeting.

This meeting is being conducted in accordance with the provisions of the Federal Advisory Committee Act. Dr. John T. Larkins the Designated Federal Official for the initial portion of the meeting.

We have received several written comments and two requests for time to make oral statements from members of the public regarding today's session on Vermont Yankee.

A transcript of a portion of the meeting is being kept, and it is requested that the speakers use one of the microphones, identify themselves, and speak with sufficient clarity and volume so that they can be readily heard.

I will begin with some items of current interest. Dr. Medhat El-Zeftamy, who has been with the agency for the past 27 years, 22 of which were with the ACRS, is retiring on January 3rd, 2006. Even though it will be good for Med, this will be a major loss for the ACRS.

On behalf of the committee, I would like to thank Med for his outstanding technical support to the ACRS in its review of numerous matters. Some of these were a first of a kind, such as the development of the license renewal process and the first license renewal application for the Oconee plant, and some of the others included the design certifications of ABWR and AP-1000, preapplication reviews of ESBWR and ACR-700 designs, policy issues related to the licensing of future plant designs, early site permit applications, NRC Safety Research Program report to the Commission for which he helped me, reactor fuels, human factors, and safety culture.

Thank you very much, Med. We wish you well in your future endeavors, and also I would add in your future relaxation. Thank you, Med.

(Applause.)

CHAIRMAN WALLIS: After the discussion, presentation, and questioning about Vermont Yankee, which I would like to inform the committee will be broadcast by telephone, we are invited to go to the cafeteria to participate in Med's retirement party.

In the items of interest handout, you will note that there's some remarks by the three Commissioners, the first three items.

I'd now like to proceed with the meeting and the first item on the agenda is the request from Entergy regarding Vermont Yankee, and I turn to my colleague, Rich.

MEMBER DENNING: Thank you.

Today because of time constraints, we are going to have presentations on just two of the critical issues, the integrity of steam dryers and the containment over pressure credit. These two topics were selected by the subcommittee because of high interest in these areas and also the feeling that we needed some more information in these areas to support the committee's review.

There are a number of other issues that have been considered that the committee will need to deliberate. These include the adequacy of the engineering inspection that was performed; the need for large transient tests; reduced operator response times; the GE nuclear and thermal hydraulic analysis methods; flow accelerated corrosion; and PRA results and application.

These areas are not cut and dry, but the subcommittee feels that we've received adequate information from the staff and the applicant to support the committee's deliberations in these areas.

I don't discourage the committee members from raising questions related to these areas if they would like, but because of the time constraints, we'll want to keep those discussions bounded.

Now, I see that George isn't here, and that may help us considerably in that regard.

(Laughter.)

MEMBER DENNING: We also have two presentations that are planned by the public, and I will ask those speakers to limit their presentations to five minutes. The first set of presentations relates to dryer integrity, and I would like to ask Mr. Thayer from Vermont Yankee to introduce that topic.

MR. THAYER: Good afternoon, Mr. Chairman, members of the committee. My name is Jay Thayer. I'm a Vice President at Entergy, Vermont Yankee.

And before we start this morning, I have one brief message for you. I would like to thank you for your deliberations on the Vermont Yankee extended power uprate over the last four subcommittee meetings. I've been impressed with the diligence and the rigor of the discussions with the committee and also the thoughtful questions coming from the subcommittee.

One message I want to leave you with this afternoon is that the men and women of Vermont Yankee and of Entergy Nuclear, for that matter, are fully committee to this power uprate, and the message I want to leave you is that we are committed to the continued safe operation of that plant, and if this uprate is granted, that commitment will not change, nor will our focus on safety be distracted for any reason. And I want to make sure you heard that from me as a responsible person for Vermont Yankee

With that, I'd like to turn it over to Mr. Brian Hobbs who will lead the presentation that the Chairman mentioned on our dryer.

Thank you very much.

MR. HOBBS: My name is Brian Hobbs. I'm the Entergy supervisor, Engineering Analyses for Vermont Yankee extended power uprate project.

This afternoon, along with Mr. Enrico

Betti on my left, I'm presenting a summary of key points from last week's subcommittee meeting regarding Entergy's evaluation of the Vermont Yankee steam dryer structural integrity.

These key points are: acoustic loads are the primary source of industry dryer degradation experience. Higher steam flows at power uprate conditions can exacerbate acoustic loads.

Secondly, Vermont Yankee's measurement configuration is capable of detecting acoustic loads that affect the dryer.

And the third point specific to Vermont Yankee, we have measured current loads and know there is some acoustic energy caused by turbulence. There is no evidence of significant acoustic resonance. The Vermont Yankee dryer structural analysis shows substantial margin to the applicable ASME fatigue stress limit. A complete exterior and interior baseline inspection and follow-up inspection of the dryer shows no preexisting structural vulnerabilities.

The Vermont Yankee dryer has been modified to strengthen it for operation at EPU conditions, and Entergy will control power ascension to EPU conditions using a dryer monitoring plan that insures dryer structural integrity is maintained.

So in summary, the key points, Vermont Yankee does not have significant acoustic loads at current power levels. We have a measurement system that can detect acoustic loads and the onset of resonance if it occurs during power ascension to EPU operating conditions.

And finally, the Vermont Yankee dryer structural integrity analysis demonstrates substantial margin to the ASME fatigue limit which will be monitored to insure structural integrity at EPU operating conditions.

The VY dryer structural analysis relies on obtaining fluctuating pressure measurements on the main steam piping. We upgraded our flow induce vibration detection capability during the recent Vermont Yankee refueling outage by installing a second generation measurement system consisting of six strain gauges at two locations on each main steam line and enhancing the data acquisition system.

We also monitor piping vibration using 21 accelerometers on the main steam piping. Baseline strain gauge and accelerometer measurements indicate that Vermont Yankee has very low vibration levels at current license thermal power, as you will see in the next slide.

As discussed in last week's subcommittee meeting, we performed an evaluation of main steam branch lines for potential acoustic excitation and concluded that some cavities may resonate at both current licensed thermal power and EPU operating conditions, but there is currently no evidence of such resonance.

This indicates that these sources at Vermont Yankee do not couple with other system modes resulting in a low magnitude response.

We also discussed how the onset of resonance would be detected via the dryer power ascension monitoring plant. Data from Vermont Yankee, the Quad Cities plant, and scale model testing indicates that excitation of acoustic sources, whether inside the reactor steam dome or in the main steam lines will be detectable in the Vermont Yankee strain gauge and accelerometer locations.

CHAIRMAN WALLIS: So what detects what's going on inside the dryer itself? You said it would detect excitation from the main steam line in the reactor vessel. What will detect what's happening in the dryer itself?

MR. HOBBS: We have data from the Quad Cities instrumented dryer, which earlier this year installed the dryer with instrumentation on the face of the dryer itself, and that data was compared to strain gauge data at the Quad Cities plant, and it was shown that whether caused by turbulence or vortex shedding inside the vessel and on the dryer, or by excitation of a cavity in the main steam lines, that any acoustic excitation could be detected in the strain gauges on the main steam lines just outboard of the main steam nozzles.

MEMBER DENNING: But in that case, the steam line itself is resonating with the dryer, and our concern is suppose, different from what apparently happened to Quad Cities, if there's some other mode of excitation that causes vibrations within the steam dryer that does not excited the steam line, can we be convinced that the signal will propagate from the dryer region into the steam line sufficiently that you'd be able to measure it there?

MR. HOBBS: Yes, we believe, and the NRC staff, I think, has also done work on this, that the vibration that's occurring in the vessel itself and on the steam dryer that's high enough to cause challenges to the structural integrity of the dryer will be detected on the measurement system on the main steam lines itself.

MEMBER DENNING: But you don't have any directly relevant analysis to convince us of that?

MR. HOBBS: Well, again, we have empirical data from the Quad Cities instrument ed dryer that shows that in their case, they were able to detect the excitation within the vessel on the steam lines.

We also have scale model test data from GE that shows that they were able to detect an excitation source inside the scale model reactor vessel on the main steam lines so that there's evidence that you can detect it, and at levels that, again, challenge the integrity of the steam dryer. That would be the case.

This curve here shows recent Vermont Yankee strain gauge data on main steam line C seven feet outboard of the main steam nozzle. This plot, which is representative of the eight main steam line monitoring locations at Vermont Yankee, shows that the small amount of energy in the Vermont Yankee system is generally below 70 hertz.

The peaks at 20, 35, 45, and 60 hertz are caused by turbulent excitation with the latter three coinciding with reactor steam dome acoustic modes. The lack of energy at frequencies above 80 hertz demonstrates suitability with the Vermont Yankee dryer modification which shifted the frequency of the dryer front hood from 20 hertz to 80 hertz.

This yellow curve is the Quad Cities' strain gauge data at original licensed thermal power at the same measurement location as on the Vermont Yankee steam line. Quad Cities, as you can see, has high acoustic energy in the 150 to 170 hertz range. This is thought to be caused by coupled excitation of the Quad Cities main steam relief valves. The Quad Cities rated main steam velocity at original licensed thermal power is approximately the same as the Vermont Yankee steam velocity at full EPU operating conditions.

The red curve is Quad Cities data at EPU conditions. The high frequency peak grew sufficiently at the higher steam flow rates to cause the damage to the front hood plates of the steam dryer and looking at the linear version of this same plot, it's evident that power uprate exacerbated the original licensed thermal power flow induced vibration phenomena at Quad Cities.

As described in last week's meeting, we used two methods to develop plant specific dryer loads at Vermont Yankee, an acoustic circuit model with a computational fluid dynamics model as well. The acoustics circuit model uses time history inputs from main steam system fluctuating pressure measurements and projects those loads onto the dryer.

Quad Cities, as I mentioned previously installed new dryers in both units earlier this year. The first of those dryers was instrumented to measure pressures and stresses acting on the dryer. The data obtained from these measurements was used to benchmark the acoustics circuit model that's applied to Vermont Yankee and allowed us to determine what the model's uncertainty was and factor it into our prior load definition.

Entergy also developed a computational fluid dynamics model which provided an understanding of turbulent vortex shedding phenomenon in the reactor steam dome. The CFD model analyzed conditions at both 100 percent and 120 percent power levels with both loads run through our structural analysis.

The results indicate that turbulent forces act primarily on dryer locations adjacent to the main steam nozzles and have little structural impact on dryer components.

In addition, the use of a compressible fluid in our CFD model resulted in the prediction of acoustic modes above 25 hertz which are similar to those we measure in our strain gauge data. Acoustic loads are detected in the latest strain gauge data at current license thermal power at Vermont Yankee.

At the subcommittee meeting last week, we discussed development and use of a finite element model on the Vermont Yankee dryer using ANSIS methodology. The CFD model and acoustic circuit model pressure time history loads were run separately through the finite element model and resulting stresses combined. The maximum fluctuating pressure at each frequency for either the 120 percent or 100 percent CFD model loads were used for the stress analysis.

The peak alternating stress calculated by the finite element model was compared to the fatigue limits in the ASME boiler and pressure vessel code. The results are shown here for the most limiting component, which is the Vermont Yankee dryer weld at the top of the vertical face. The peak calculated stress of 5,450 psi combines the acoustic circuit model and CFD model loads and includes weld geometry and stress intensification factors.

The acceptance limit is the ASME fatigue curve C limit of 13,600 psi. Our limit of power ascension at Vermont Yankee is 7,400 psi, which gives us a margin for uncertainty in our structural analysis. Applying the limiting component margin to the stress limit and incorporating uncertainty, we calculate a limit curve factor to be applied during power ascension.

CHAIRMAN WALLIS: Could I understand how you came up with 7,500 psi?

MR. HOBBS: For our limit we can show you how we did that, and I'll ask Mr. Betti to help out here.

MEMBER DENNING: Definitely speak into the mic and also introduce yourself.

MR. BETTI: Enrico Betti, Entergy.

The 7,400 is based on the sum of the squares combination of 18,000, 1,850, and 5,124 from the two analyses that we ran. With the addition that we applied the limit curve factor that we're applying in our start-up curve times the ACM number. So it's the 1,857 times the 2.87; that quantity squared plus 5,124 squared, the square root of the combination of that.

When we developed our limit curve factor in uncertainties, we actually worked from the LCF equation you see right here, and then that's also how we developed our uncertainties based on the CFD and ACM uncertainties.

Another interesting point to make here is even though the CFD value is larger here, we did quite a lot of evaluating of the CFD analysis, and of that 5,000 stress, about 1,000 is due to the turbulent forces and 4,000 is due to acoustic forces that were just a byproduct of the compressible gas modeling we used.

So it is a double dipping that we're accounting acoustic forces, and we really meant the CFD only to give us the effect of the turbulence in our model, but we included both in this analysis.

CHAIRMAN WALLIS: So the acoustics are counted twice.

MR. BETTI: Acoustics is counted twice. It has a big factor here --

CHAIRMAN WALLIS: I would be a little happier if you had shown there was no shaking. What you're showing here is there is shaking, but it's almost a factor of two below some limit in terms of limiting stress. But there still is shaking. You're not saying that it's not going to shake. It's going to shake, but not shake apart is what you're saying.

MR. BETTI: I think I'd like to clarify that a little. From our instrumentation system that we --

VICE CHAIRMAN SHACK: Don't go away yet.

MR. BETTI: From the instrumentation system that we have there, our calculated shaking stresses our under 2,000 psi peak stress, and that's using a conservative stress intensification factor.

We have a CFD model that wasn't designed to do acoustics. So we don't have proper dampening, et cetera, but as a byproduct of compressibility, we calculated a higher amplitude than we see on --

CHAIRMAN WALLIS: The thing that concerns me with CFD and acoustics is that once the thing starts to shake, it feeds back to the acoustics, and your CFD doesn't consider a moving boundary, does it? CFD is just rigid boundaries. You calculate the fluid stress, and then you let it shake the object, but you don't feed back the shaking of the object to the fluid mechanics, which actually gets things really going if they're in tune with each other.

That's missing, isn't it, here?

MR. HOBBS: Dr. Wallis, if you remember the blue curve from our strain gauge measurements, there's almost no energy above 80 hertz. So the --

CHAIRMAN WALLIS: That's reassuring, yes.

MR. HOBBS: -- the energy that's reflected here is a prediction from a CFD model that has some uncertainty associated with it, and this is for a 120 percent flow case.

So what we're doing is we're projecting our CFD load from our model to the 120 percent case and applying that to today's load definition. So we have the computer shaking, a small amount of shaking going on that we don't reflect in our actual plant measurements, but may occur at power uprate conditions, and that's why we have a monitoring program.

CHAIRMAN WALLIS: So you're saying that these values are much bigger than you'd get from your actual measurement.

MR. HOBBS: Yes.

MR. BETTI: Yeah, our measurement value is the 1,857, and that's a peak value.

CHAIRMAN WALLIS: But you understand what I'm saying about fluid structure and direction. I don't think we're yet smart enough to put in the CFD and the motion of the boundary.

MR. BETTI: We agree.

CHAIRMAN WALLIS: It would be good if you could.

VICE CHAIRMAN SHACK: I mean, you're computing your load factor on just the acoustic mode stress. I mean, what gives you such confidence that you can take your limit stress and just subtract off the CFD? You know, this somehow seems as though you're assuming that the CFD stresses are real. You know those with minimal amount of uncertainty, and for the reasons that Dr. Wallis has talked about, I'm not sure why you don't consider them at least as uncertain as the acoustic mode stresses.

MR. HOBBS: Right.

VICE CHAIRMAN SHACK: And yet you're not doing that here. You're showing a load factor as though those were the exact stresses, and all of my uncertainties are just dumped on the acoustic mode.

MR. HOBBS: Right. Well, there's two uncertainties we show here. One is the 3.91, and that is the total uncertainty from both our acoustic circuit model and our CFD analysis. What we are basing the limit curve factor on is how much growth can we tolerate for acoustic loads as we increase to power uprate conditions, and the reason we hold this CFD loads as being a Row B squared type load is because we don't think the turbulent CFD loads are going to increase with as much potential for residents as the acoustic loads at EPU conditions.

So this is kind of the head space. The

2.87 factor is how much can we tolerate in the way of increasing acoustic loads. That does not give you the uncertainty number for the analysis.

CHAIRMAN WALLIS: Now, what surprised me a bit is when we look at your red, blue, and yellow curves, the Quad Cities values are four orders of magnitude above yours. So I mean, here you're talking about a factor of two, you know, in your previous slide.

MR. HOBBS: Right.

CHAIRMAN WALLIS: Your 13,000 and 7,000, but here you're talking about a factor of 10,000. Now, I can believe that you're much better than Quad Cities experimentally. Why aren't you so much better when you calculate things?

MR. HOBBS: I think, Dr. Wallis, the results are similar actually. I think that when we take a CFD analysis , that gives us a localized street. If you don't look at the CFD acoustic effects, which really weren't tried to be modeled correctly, and we don't measure those high amplitude of bumps that the CFD created, that our actual measured hydrodynamic stress on a model was on the order of a couple hundred psi, and then we conservatively multiplied that times -- because we have so much margin, we use stress concentration factor and conservative weld geometry factor of five. So we take a stress that's a couple hundred. We turn it into 1,000, and then we're including after conversation with the staff this summer the -- we're keeping in the acoustic portion of that load, not filtering it back out, even though we're measuring the acoustic loads in our piping and we're using our acoustic model to project those back on our vessel.

Now, what they've done at Exelon is only look at the acoustic portion of the load. So I think this is a very conservative picture of the street state at Vermont Yankee.

MR. HOBBS: This demonstrates how we'll apply our limit curve factor during a power ascension. Recall that on this curve here, the Vermont Yankee measured strain gauge data is the blue line. If we apply the limit curve factor of 2.87 to this spectra, then what appears is the green line here.

And the green line is the limit curve that will be applied during power ascension to assure that the Vermont Yankee --

CHAIRMAN WALLIS: The green line is a conservative version of the blue line. Is that what I understand?

MR. HOBBS: It's the blue line times 2.87.

CHAIRMAN WALLIS: Raised up by a factor.

MR. HOBBS: Which is our head space for incurring acoustic residence at EPU operating conditions.

Note that the Quad Cities original and EPU acoustic peaks exceed the Vermont Yankee limit curve. If the VY limit curve is challenged during power ascension, we will evaluate to insure continued acceptable dryer performance for maintaining structural integrity.

CHAIRMAN WALLIS: That's as long as nothing shakes at 60 hertz. Sixty hertz, it's a minimum for Quad Cities and a maximum for you.

MR. HOBBS: Right, and there's some --

CHAIRMAN WALLIS: Your conclusions you just drew are up in the 100 hertz and above region.

MR. HOBBS: That's right, and that's the reason, again, that's caused by coupled resonance in the main steam lines at Quad Cities. Vermont Yankee has only one relief valve in each steam line, has only one safety valve in each steam line. Quad Cities has multiple safety valves and relief valves in each steam line. So that's how the coupling occurs, because they're in close proximity to each other.

CHAIRMAN WALLIS: You measured the mechanical resonances in this steam dryer?

MR. HOBBS: Yes.

CHAIRMAN WALLIS: You hit it and see how it rings, that kind of thing?

MR. BETTI: No, we evaluate the steam dryer with answers.

CHAIRMAN WALLIS: It's all calculation.

MR. BETTI: All calculation, and what we do is we --

CHAIRMAN WALLIS: What sort of range of resonance frequencies do you find?

MR. BETTI: For where these acoustics began, with the turbulent load back, is the front face of the dryer. Brian had mentioned earlier that the fundamental frequency of the front face is around 85 hertz.

CHAIRMAN WALLIS: Eighty-five.

MR. BETTI: And that's based on the modification that we did. We --

CHAIRMAN WALLIS: You stiffened it up and braced --

MR. BETTI: We stiffened it up, yeah. Based on GE's review of a lot of reactor data, the bumps that we see in our strain gauge data at these residencies and at this 20 hertz frequency are pretty typical, not typical of Quad Cities to see these very high frequency loads, but for the data that GE had, they based the design that brought that vertical face frequency above their experience base of reactor data, and that's the modification that we have in place.

So, I mean, for us when we watch this data, of course, we have to be very sensitive to anything to show up around our fundamental frequency of our front fix. That would be a very sensitive area.

MEMBER RANSOM: What are the units of the ordinate?

MR. BETTI: The units of the ordinate are micro strain squared per hertz.

MEMBER RANSOM: Strains in -- strains?

MR. BETTI: It's strain, and the correlation between micro strain and psi -- no micro strain and psi is -- the correlation is approximately 3.9 psi per micro stain is the conversion for the VY main steam piping. It's 18 inch, Schedule 80, and we did do UT data on the piping when we installed our strain gauges so that we would have an accurate assessment.

MEMBER RANSOM: Psi seconds, I guess.

You've got frequency and per second, right?

MR. BETTI: It's power spectral density. So we take the FFT response times conjugate. Times conjugate, right, and then divide that by -- it's shown as the per unit hertz. It's a way to normalize it so that there wasn't a question on how you normalized your curves. If we use PSD, power spectral density, it's more of a uniform way that we could find it doesn't make it subject to how something normalized your FFTs, magnitude.

MEMBER DENNING: Let's continue because there are only a few more slides, and if we have some other questions we can come back to them.

MR. HOBBS: Okay. The Vermont Yankee dryer power ascension monitoring will include power increased steps and test plateaus at each five percent of current licensed thermal power. Data will be collected hourly when power is increasing and within one hour of reaching each test plateau.

In accordance with the NRC license condition if the limit curve is exceeded, power will be reduced to a previously acceptable level within two hours and an engineering evaluation performed to document continued dryer structural integrity.

Also in accordance with the NRC license

condition each test plateau has a 24-hour minimum hold time to collect strain gauge, accelerometer, and plant data, perform inspections, and evaluate results.

The evaluation will be provided to the NRC staff and power will not be increased until 96 hours after confirmation of receipt by NRR.

In conclusion, the Vermont Yankee dryer structural integrity evaluation demonstrates that the VY dryer shows no significant vulnerability to flow induced vibration at current licensed thermal power; utilizes a methodology that can detect significant acoustic excitation either in the main steam lines or reactor steam dome; and finally, demonstrates ample margin to the code allowable fatigue limit which will be monitored during power ascension to insure dryer structural integrity is maintained.

MEMBER DENNING: Would you comment on a couple of things for me? One of them is on the cracks that have been observed, could you give a quick review for the committee members that weren't here as to what your perception is, the origin of the cracks, what you've done with those cracks that you've repaired, what you believe the origin is of the cracks that have not been repaired?

MR. HOBBS: Certainly. The Vermont Yankee dryer was inspected for the first time in 2004 and a complete internal and external inspection was performed at that time in accordance with Silth 644 from General Electric. What we found in 2004 were a total of 20 indications. Two of those were located in the steam dam area of the dryer. Those were repaired and the cracks ground out. Those cracks were determined to be caused by fatigue, and it was thought that they were created originally due to construction of the dryer because they were 180 degrees out from each other, and they grew to a length of about three inches, and we concluded GE also analyzed this, that it was cold spring that caused those cracks. So they were ground out and repaired.

Two other cracks were found in the drain channel and drain pipe areas of the steam dryer, which is in the skirt where the water drains from the dryer vein banks and down the skirt and back into the reactor region. These two cracks were on the order of 14 inches or less in length. They were determined to be caused by IGSEE based on their location and their characteristics, and those were left as is. Those were not repaired.

There were 16 other indications found on the dryer vein banks, and the vein banks are in a low stress area of the dryer. They basically hold the veins that remove the moisture in the dryer, and they're sort of the frame around the vein dryer banks, and those were all inch and a half or less in size and were thought to be caused by IGSCC, although there may have been some fatigue involved in those cracks as well.

The characterization of those is they were very tight indications, and we did an analysis that said even if those indications grew to the entire length or to the entire width of the vein and bank, they would still be structurally intact. So those were not repaired also.

In 2005, last month we went back in and did an inspection of all the indications we found, and we also inspected the repaired indications from 2004, and we found that there was no growth in the dryer drain channel or drain pipe IGSCC indications. Those had not grown in size, and we also checked the previous indications on the dryer vein banks and found that those also had not grown in size, but we did find additional indications on the vein banks, and again that's the areas that are on the edges of the dryer vein banks.

The reason that we found additional

indications and the total additional indications we found were 46, is because we used an enhanced digital inspection system for this outage whereas previously we had used an analogue inspection system with a VCR type videotape. This time we used digital media, and the resolution was much better.

So it's essentially an enhanced visual inspection we did finding more indications similar to those we had previously found.

The characterization was tight tracks, and again, if they grew to the entire width of the dryer vein bank and plate, they'd still be structurally intact.

MEMBER DENNING: And you have a commitment after power uprate to inspect the next three outages, correct?

MR. HOBBS: That is correct, and it's a thorough internal and external inspection for the three refueling outages.

MEMBER DENNING: this is the only presentation we're going to have on the integrity of the dryer. So are there any other questions you'd like to raise now?

VICE CHAIRMAN SHACK: I mean, we're talking about carbon steels here. You're saying

IGSCC.

MR. HOBBS: Stainless steel, Dr. Shack.

VICE CHAIRMAN SHACK: Stainless steel.

MR. HOBBS: Yes.

MEMBER DENNING: Okay. Are there any other questions?

VICE CHAIRMAN SHACK: One more question.

MEMBER DENNING: I'm sorry.

VICE CHAIRMAN SHACK: Is there any consideration that your fatigue stress limit, which in the code is based on air data, will be lower in the environment?

MR. HOBBS: That's a good question. Mr. Betti, can you help me out here?

MR. BETTI: No, I wouldn't be the best person to ask.

MR. HOBBS: Can we get back to you on that question? So the question is would the limit be lower in a --

VICE CHAIRMAN SHACK: Like say a factor of two.

MR. HOBBS: Okay.

MEMBER POWERS: Isn't there a relatively famous publication by Dr. Shack on that subject?

VICE CHAIRMAN SHACK: Well, such effects

are known to occur. Now, whether they're particularly applicable in this case is another question, but it does seem like an issue that ought to be addressed.

MR. HOBBS: So two phase or liquid environment versus air.

CHAIRMAN WALLIS: Or a changing liquid and vapor environment.

MEMBER DENNING: If you could get back to Mr. Caruso with any comments by tomorrow, is that reasonable?

MR. HOBBS: Certainly.

MEMBER DENNING: Thank you.

MEMBER POWERS: We're not going to put any pressure on you.

MEMBER DENNING: Not going to put any pressure on you. Thanks.

Now we're going to switch to containment over pressure credit, and we're going to have presentations that relate to a PRA analysis of what's the significance, and then we're going to also have a presentation on what are the conservatisms and if you do a realistic analysis what happens.

And the first presentation is going to be by Mr. Stutzke on the PRA.

CHAIRMAN WALLIS: Now, we have two

presentations. You say one is PRA. One is conservative. I had great difficulty figuring out in the PRA whether there were conservatisms incorporated or not and how things like uncertainties were handled because if the conservative method shows there's no problem, how can the PRA possibly reveal there is a problem?

I mean, you're going to explain all of that to me?

MEMBER DENNING: Rick, did you want to make any introductory remarks?

MR. ENNIS: My name is Rick Ennis, and I'm the project manager in NRC's Office of NRR for the Vermont Yankee extended power uprate, and the two presentations that we're going to present today regarding containment overpressure credit are a risk evaluation of the proposed crediting by Marty Stutzke and then the deterministic evaluation by Rich Lobel.

MR. STUTZKE: Hi. I'm Marty Stutzke, a senior reliability and risk analyst in the Office of Nuclear Reactor Regulation, and I'm here today to discuss Entergy's risk evaluation of the proposed credit for containment accident pressure to provide net positive suction head to the low pressure injector or coolant injection and the core spray pumps.

Briefly stated, Entergy has completed its risk evaluation. I've reviewed the information that they've submitted which confirms the conclusions that are present in the current draft safety evaluation. So we're in the process now of supplementing our safety evaluation to reflect the additional information that Entergy has provided in supplements. I believe it's 38, 39, and 43, totaling some about 400 pages of information.

The second point is using the realistic assumptions to estimate --

VICE CHAIRMAN SHACK: Excuse me. Is Supplement 43 posted somewhere on the Web? Can I get it? I don't believe I have it.

PARTICIPANT: Actually I think it's in the package.

PARTICIPANT: It was received December 2nd.

PARTICIPANT: Yes.

MEMBER DENNING: Before you go on, would you comment on maybe -- Mike may be the more appropriate one to talk to -- but we have been concerned about the fact that we did not have a final SER, and we were told that the subcommittee that you were looking at this as a confirmatory kind of request. Is that still your comment? Is that still the staff's position that basically the essence of the SER is unchanged?

MR. STUTZKE: That's correct. I have not changed my conclusions based on what Entergy has provided recently.

So Mr. Lobel will talk about some additional insights on the calculation of available NPSH, the margins available, this sort of information.

CHAIRMAN WALLIS: Will you accept questions on this slide now? I had great trouble because your second bullet says if you -- I think it's saying if you calculate the pool temperature, suppression pool temperature, realistically you don't need CAP credit.

Now, Mr. Lobel told us that if you calculates conservatively, less conservatively than is required by the design basis assumptions, but you still calculate it conservatively. You don't need CAP credit. So how can you possibly have any effect on risk if you don't need it? How can NPSH -- if the problem never arises, how can it ever affect risk?

MR. STUTZKE: It can't.

CHAIRMAN WALLIS: Well, so why are you doing risk analysis that shows there is a risk? It doesn't make sense?

MR. STUTZKE: Well, the risk analysis that I did, it's a "what if" analysis.

CHAIRMAN WALLIS: You mean what if two is not true anymore?

MR. STUTZKE: That's correct.

CHAIRMAN WALLIS: Well, that's silly because risk is supposed to be realistic analysis, isn't it?

MR. STUTZKE: Well, I don't know if it's silly or not. We prefer to call it epistemic uncertainty, I think.

CHAIRMAN WALLIS: I think it's more regulatory assumption uncertainty, isn't it? You shall make an assumption which is not realistic and then look at what would happen if you did that. Isn't that what you're doing?

MR. STUTZKE: That's correct.

MEMBER DENNING: Well, Marty, how strongly do you feel -- how confident are you in that second statement about the realistic assumptions indicate that the overpressure credit isn't necessary?

MR. STUTZKE: As confident as I can be without actually doing the experiment, which I hope that we would never do like that.

MR. LOBEL: I'm going to get into that and show some calculations in some detail. So you'll see the assumptions that go into that statement.

MEMBER DENNING: Okay. Very good. Well, at least we understand the ground rules then of what the risk analysis is showing, which is it says a "what if." We're getting into the --

CHAIRMAN WALLIS: You're going to clearly explain what the "ifs" are.

MR. STUTZKE: Right, and I have some additional information on that later on to try to explain.

Okay. The last thing is that we have compared the proposed containment accident pressure credit to the five key principles of risk informed decision making in Reg. Guide 1.174, and I'll discuss how the insights from the risk evaluation support the conclusion later.

Okay. I've continued my chronology that I have provided to the subcommittee in various meetings in December of how we go into doing the risk evaluation here. I think what's important is what's new since the last subcommittee here is that Entergy has responded formally to the request for additional information I sent on their supplements 38 and 39.

That was done on Friday about noon as I remember, which totally spoiled my weekend, but that's how recent the information is here.

One thing I should point out, too, is Entergy's evaluation is independent of mine. In other words, I get to ask them questions, but they don't get to ask me questions on what I did and why I did it like that. So let me assure you they've not seen my actual PRA model or any of the calculations that it has produced. This is their own work.

I would also point out that they basically completed their evaluation before any of the subcommittee meetings we had. So they didn't even have the benefit of my results to drive them there. So it's about as independent an analysis, I think, as could be construed.

But let me talk to Dr. Wallis' question a little bit more. What we're dealing with here is that the proposed accident pressure credit introduced a modeling uncertainty into the PRA. In other words, we have success criteria for the PRA and the success criteria says that we don't need containment integrity in order to insure net positive suction head to the pumps, and the success criteria are based on realistic estimates of available NPSH. Okay?

But we know these estimates are uncertain. They're so-called phenomenological uncertainties with them. What are the friction factors? What's the containment response and hence its pressure and so forth and so on, and as a result, the success criteria used in the baseline PRA are uncertain.

That's a type of modeling uncertainty, and the accepted way of attacking that type of modeling uncertainty to get to the bottom of it is to do what's called sensitivity analysis on this. And specifically what people do then is to propose an alternative set of success criteria. In other words, in the alternative set, we would just assume the pressure credit is necessary. In other words, the failure of the containment's integrity actually gets us into trouble with no positive suction head on the pumps.

CHAIRMAN WALLIS: So you're assuming something which someone else has shown to be impossible.

MR. STUTZKE: No necessarily.

CHAIRMAN WALLIS: I thought Rich was going to show it was impossible. It's going to be so conservative that it could never happen.

MR. STUTZKE: I'll say there's always the uncertainty involved here.

CHAIRMAN WALLIS: Well, I thought a bounding analysis or a limiting analysis using something like the first law of thermodynamics gave you pretty much the certainty, but maybe we haven't got that far. I just want to be sure how certain he is about it will never happen.

MR. STUTZKE: Well, the way that I look at the sensitivity study is we do two cases, one assuming no credit is needed and one assuming that credit is needed, and so the truth is somewhere in between those two numbers.

CHAIRMAN WALLIS: But it's a huge leap to say that something which you know is almost never going to happen actually is needed. So you really should downgrade your numbers you've got at the end because of that.

MR. STUTZKE: Yes.

CHAIRMAN WALLIS: Is that why they get a number which is much smaller than yours?

MR. STUTZKE: Actually the number is higher than mine.

CHAIRMAN WALLIS: Theirs is higher than yours.

MR. STUTZKE: Right, and I tried --

CHAIRMAN WALLIS: Ah.

MR. STUTZKE: I have a slide on that, about why that is.

CHAIRMAN WALLIS: Okay.

MR. STUTZKE: Okay? Okay. Let's jump to the next slide.

It took me some time to understand why they got different results. Realize we're in the realm of a sensitivity study. So different analysts would tend to make different assumptions trying to get at this.

But the difference between Entergy's approach and my approach seems to boil down to two main differences. Okay? One is they use different success criteria than I did. The scenario is this.

CHAIRMAN WALLIS: You mean when you do a PRA you can arbitrarily choose what you want?

MR. STUTZKE: Of course.

MEMBER POWERS: It is arbitrary.

MR. STUTZKE: Well, it does have basis. I can explain why I did what I did, okay, in a little bit. Let me get down the slide here.

First of all, they credit alternative injection sources, and I didn't. These alternative injection sources, for example, for medium size LOCAs, they considered condensate, control rod drive system, condensate transfer for transients and small LOCAs.

In addition to that, they included feedwater, HPCI, RCIC, these sorts of things. For large LOCAs, there's even a consideration of fire water injections and interconnection with the service water system and RHR loop alpha.

I gave no credit at all to alternative injection sources in my risk assessment. The reason, to be honest, is I was trying to save myself some work. As you see, on the second line there, I had credited; I had focused my attention on suppression pool cooling following loss of containment integrity. That was the notion that even if containment integrity is lost early, it takes time to heat up the large mass of water in the pool, and if the operator got suppression pool cooling up and running in time, it didn't matter that he had lost integrity.

Okay. How does that save me work? Well, the answer has to do with human reliability. The fact is a dependency among the operator actions to start suppression pool cooling. It's a manual action, and line up alternative injection sources. Okay?

And the nature of the dependency involves the cognitive error. If he misses the scenario, he's not likely to do any of these things. He won't understand what's going on here.

The assessment of dependency between or among various operator actions is rather involved, and I tried to save the work by just not crediting the alternative injection source whatsoever.

Okay. The second thing and perhaps more puzzling here is the difference in the presumed probability of preexisting leakage into the containment. You'll see that Entergy's estimate is almost two orders of magnitude below mine, and so I had to question. In fact, that was the basis of one of the RAIs that I had asked why do you get this large number.

There's several things going on here. First of all, Entergy picked a break size or a containment leakage size of 60 L sub A. It's the definition of failure of the containment in their sensitivity analysis, whereas I had picked 35 L sub A.

Okay. Realistically, I guess it's infinity times L sub A. Okay? So we have to pick some sort of break size and assign a probability to that number or to that break size like this.

CHAIRMAN WALLIS: I see. So yours is smaller, therefore, more likely. Is that it? That's the tendency that you would expect.

CHAIRMAN WALLIS: Why does this have to be preexisting? I mean, couldn't the actual high pressure during the beginning of the LOCA cause a leak which then causes the depressurization later on?

MR. STUTZKE: Well, it could, but the probabilities related to the time between tests, time between when you know the containment is actually intact. So the mission time of the PRA is small we consider as compared to the preexisting --

CHAIRMAN WALLIS: But you didn't consider leaks caused by the accident itself.

MR. STUTZKE: But not phenomenological leaks.

MEMBER DENNING: But I think our belief would be that would be a very small probability.

CHAIRMAN WALLIS: Well, how small is small?

MEMBER DENNING: Certainly smaller than ten to the minus two probability.

MR. STUTZKE: I mean, what you're asking is if you pump the containment up to a few PSI, will you explode it.

CHAIRMAN WALLIS: Oh, no, no. Will some small place get proper leak, not a real big failure, but just a little hole?

MEMBER DENNING: Still our evidence is that containments can take like two to three times the design pressure without introducing those kinds of --

CHAIRMAN WALLIS: With nothing connected to the containment?

MEMBER POWERS: Has a containment failure probability analysis done on this containment?

MEMBER DENNING: On this containment? Probably not on this one, but very similar.

You meant for --

MEMBER POWERS: Yeah, the usual number of two to three is quoting from some test rests.

MEMBER DENNING: Well, I think they really preceded the test results, the two to three. The test results have been confirmatory or indicated those are pretty conservative, I think.

MEMBER POWERS: When we calculate containment vulnerabilities, whatnot, we find that they're very, very design specific. When we experiment with them, we always find they fail at flaws. They're not usually in the models.

MEMBER DENNING: But still well above the design basis. I don't know any evidence of tests that we've done that would indicate that containment would fail, have a significant leakage as a result of this kind of a pressure pulse.

MEMBER POWERS: No, it would be something unexpected. It would usually, for these kinds of containments, it will be a seal failure having nothing to do with pressurization or whatnot. The principle issue with all of these things is none of the analysis take into account construction flaws.

PARTICIPANT: But that's why you do the tests.

MEMBER DENNING: Continue.

MR. STUTZKE: Well, I should point out that the basis of Entergy's containment failure probability is a rather new EPRI report. It's based on expert elicitation. The staff is in the process of reviewing this report. It's being submitted in the context of granting permanent 15-year ILRT extensions.

Okay. But the staff has, in fact, made a number of comments on this report. So we haven't accepted it or rejected it.

MEMBER POWERS: What particular thing was elicited from the experts?

MR. STUTZKE: They asked the experts to predict or estimate the probability of various containment failure modes generating various leak sizes, discrete leak sizes.

MEMBER POWERS: I'm always puzzled how you find an expert on those particular subjects since we haven't had any.

MR. STUTZKE: That's a large part of the staff's discomfort with this report.

MEMBER POWERS: I understand.

MEMBER DENNING: Now, wait a second now. But this relates to not an induced failure but a --

MR. STUTZKE: A preexisting.

MEMBER DENNING: -- a preexisting failure, and we've had plenty of those historically with ILRTs, not in recent history as much as earlier history.

MR. STUTZKE: That's correct.

At the same time, the failure probability that I had used in my study came out of NEI interim guidance on temporary changes to ILRTs. Okay? But it's actually what I'll call a data driven approach, zero failures in 182 tests. Okay? And they do their Bayesian update of this.

The difficulty with this type of data is it speaks nothing to the break size. All you know is that you passed the ILRT, yes or no. Okay?

So in some respects the newer EPRI data is a little better. It gives you a downward curve that says the bigger the hole, the lower the probability, and the question is how much do you want to believe that.

Okay. The other thing that you have to realize is when you put these lines together, you have competing effects going on here. Okay? In other words, the credit for alternative injection sources, the probability of failure seems to be higher for those than for the expression pool cooling system. At the same time the containment failure probability is lower, and it took me a while to sort through all of this to understand.

But I think I understand it in terms of the minimal cut sets and the numbers that drive the answers now.

Let's jump to the next slide here.

CHAIRMAN WALLIS: Well, these credits, presumably the whole picture is really some sort of synthesis of what they did and what you did, and then you can make various choices about do you credit this or credit that, and I would think what you have to do then is say, well, what's the probability of suppression core cooling, not just arbitrarily credited, but what's the probability of it happening?

MR. STUTZKE: No, but that's what the PRA does. It's just that I worry about --

CHAIRMAN WALLIS: But if they don't consider it at all, they can't have nay probability assigned to it except presumably --

MR. STUTZKE: Well, basing the failure probability is one. That's their default assumption.

CHAIRMAN WALLIS: But the realistic thing is to put yes everywhere and then evaluate a probability.

MR. STUTZKE: That's correct.

CHAIRMAN WALLIS: I see.

MEMBER DENNING: You never showed the bottom line.

CHAIRMAN WALLIS: Never talked about them.

MR. STUTZKE: Well, I want to talk about it, I guess, in terms of the plot on the next page. The fact is that they generate a change in CDF due to the over pressure credit assumption alone that's about an order of magnitude higher than mine. When I add that change in core damage frequency to the change in core damage frequency due to other impacts of the proposed EPU, I get a total change of about 90 minus seven per year. Okay?

And plotting that against their baseline CDF of 80 minus six per year, you end up with a black dot, which you can see it's right on the border line between Region 3 and Region 2, okay, but that's still equivalent or translated in Reg. Guide 1.174 as a very small change in risk.

CHAIRMAN WALLIS: And again, this is a hypothetical change. The real change in risk is probably much smaller than that.

MR. STUTZKE: It's much smaller than that. Okay. So you're right. It is a hypothetical change, depending on which set of success criteria you want to believe like this.

Okay. Talking a little bit about the five key principles of risk informed decision making --

VICE CHAIRMAN SHACK: The difference in the success criteria, you kept the success criteria, but actually tried to work out the actual probability that you'd use the containment overpressure. As I understand what they did, they just gave it up.

MR. STUTZKE: No, they have a probability of -- anther way to look at it is the scenario you're talking about is you have a LOCA and the containment is not intact. Okay? So that's kind of the challenge to the system, and the question is what happens following that.

In their study they say, well, we'll just presume the low head pumps are all failed and we'll go credit alternative sources with various probabilities.

CHAIRMAN WALLIS: When in reality they would not fail if we believe this conservative analyses. They would not fail.

MR. STUTZKE: That's right.

CHAIRMAN WALLIS: So they're assuming something which analysis shows you to be very unrealistic or, let's say, unrealistic.

MR. STUTZKE: That's right.

CHAIRMAN WALLIS: It's a strange way to do things. I suppose if you want to be really sure, you might as well do it. The whole idea of PRA was to be as realistic as possible.

MR. STUTZKE: Well, I would shy away from the bounding analysis. The reality is when you have a modeling uncertainty like this, which set of success criteria do you want to do, we turn to sensitivity studies, and in my opinion sensitivity study is always kind of a crap shoot. What you hope is that it's not sensitive.

CHAIRMAN WALLIS: My solution is not to do that at all, but to put the modeling uncertainty in the PRA and do it right, not to have this crap shoot.

MEMBER DENNING: But, Graham, the other

point is from the licensee's viewpoint he's going to -- what he's trying to show is it doesn't matter.

CHAIRMAN WALLIS: I understand.

MR. STUTZKE: It really doesn't matter.

CHAIRMAN WALLIS: But you create a precedent. You've done it this way and it has been accepted. Someone else will do it the same way, and it might not look so good, and what do you do then?

MEMBER DENNING: And, in fact, this is the way we really do look at the sensitivity to these modeling uncertainties rather than attempting to get into deep phenomenological details.

MR. STUTZKE: That's correct. It's unfortunate that Professor Apostolakis is not here. He has written several papers on this.

CHAIRMAN WALLIS: How about the length of time involved? I mean, this credit is taken for days, is it? Doesn't that make a difference? I mean, you're just saying that your analysis covers that all, all together. Nothing untoward happened. There would be no more probability of leak in the containment if it has lasted for a week than if it lasted for half a day or something?

There's no influence of time on the integrity of containment or any of the other assumptions?

MR. STUTZKE: No, no, because the probabilities are being driven by preexisting leaks, not the mission time following the LOCA, following the initiating --

CHAIRMAN WALLIS: So our concern in our previous letters with time is irrelevant?

MEMBER DENNING: Only if it's a real phenomenon. I mean, if it's a real phenomenon, then it's not irrelevant.

CHAIRMAN WALLIS: Would you show me an unreal phenomenon?

MEMBER DENNING: I think that's exactly what we're doing, Graham.

CHAIRMAN WALLIS: So we should forget about our concern with time? I mean, we're at a point in three or four letters, I think, about only for short times.

MEMBER DENNING: Well, again, if it's real and you really need to operate pumps in cavitation, then time makes a lot of difference.

MR. LOBEL: Can I? This is Richard Lobel of the staff.

Let me clarify a little that what we're talking about here is Vermont Yankee, and the situation requiring overpressure for a certain amount of time or it being more of a real effect may be the case for other reactors going through the same type of accident.

I think the numbers that we're showing, the numbers that I'll show are really Vermont Yankee specific, and I was going t omake that point a couple of times. So just I don't want to mislead the committee.

And another point I'll make is that we're really talking about Vermont Yankee here, and we're not talking about the Reg. Guild 1.82, and the conclusions we're drawing here are just for Vermont Yankee. So your more general concerns remain for us to answer, but in terms of Vermont Yankee, the numbers we're showing show the kind of conclusions we've been talking about

MEMBER BONACA: I have a question here. During your presentation two meetings ago, you pointed out a limiting case for which there is a need for the NPSH credit is the case where you have RHR. You assume failure of the RHR, right? I'm sorry?

MR. LOBEL: The single failure is the failure of an RHR heat exchanger.

MEMBER BONACA: And that's really the

phase for which you're entering the credit. For the other cases where you assume the single failure is failure of the containment, okay, you do not need the credit.

MR. LOBEL: Yeah, and I'll show that in my presentation.

MEMBER BONACA: Okay. The question I have is that in your PRA analysis, what do you assume? You assume that the RHR also is not working?

MR. STUTZKE: No, it includes failure of both trains, all the trains of RHR progressing to core damage.

MEMBER BONACA: Okay. That's the entergy assumption, if I can see that table before.

CHAIRMAN WALLIS: They made a bigger assumption, right? They've assumed the failure of containment leads to loss of NPSH, whereas in fact even with loss of an RHR, if you're realistic, you still don't need the NPSH credit.

MEMBER BONACA: I'm sorry. Could you repeat what you said?

CHAIRMAN WALLIS: I think I'm right in saying that they claim that if you lose RHR train and you realistically calculate the suppression pool temperature and you fail the containment, you still don't have a problem with NPSH.

MR. STUTZKE: That's correct.

CHAIRMAN WALLIS: So it's three things. That's why it's piling things on, isn't it?.

MEMBER DENNING: Marty, I'd like you to finish in ten minutes. I realize that's not totally under your control.

MR. STUTZKE: Yeah. I'll do my best, but I'm determined I'm going to present these slides because I stewed over them for a couple of months now.

Let me jump right to Slide No. 8 because I think it's one of the hearts of the matter here. When we look at the five key principles of risk informed decision making, I think there's two important things you need to bear in mind here. One is all of the principles have to be considered in reaching a decision. Okay?

Let's continue to Slide 8 here.

In other words, no individual analysis is sufficient. So in other words, we reach decisions that are not risk based, but they're risk informed like that.

But the reality is that there's an interconnectiveness among the various principles like this. I make the analogy to checks and balances in federal government between judicial and executive and things like this, and you guys have written several papers that reflect that balance and the struggle of trying to decide what the appropriate balance is here.

I have cited two of my favorites here because I love the language that was used like this, but the points here are trading off defense in depth when PRA tells you that maybe you don't need it. On the other hand, if the PRA is uncertain enough, you use defense in depth to try to compensate for that uncertainty.

So we have this balance, and that's all I'm trying to point out here, is that the issue is not what the PRA says, you know, as far as what's delta CDF, but these other factors need to be considered in here.

That being said, let me tell you how we've looked at defense in depth here. Slide No. 9 says we're consistent with defense in depth philosophy because we've met four objectives stated in the standard review plan Chapter 19, and you can read them for yourselves on there.

What I would point out here is, first of all, that the bottom line there you say overall redundancy and diversity among the varies is sufficient to insure compatibility with the risk guidelines.

So in other words, it implies if delta CDF is small enough, I must have adequate defense in depth. It's an example of the interconnectiveness among the various principles in my mind.

More importantly for this, if you look at the top three, it allows some increase in challenges to barriers or barrier failure probabilities or dependencies among barriers. That may be acceptable.

The operative word here in my mind is "significant." Okay? But the reg. guide and standard review plan are silent on what we mean by "significant," and the fact is we have to use our judgment on a case-by-case basis to decided when it's okay.

So there is a struggle in trying to decide what the appropriate balance is among these elements.

MEMBER DENNING: At the risk of destroying my plan, I do question the number two bullet there in terms of "does not significantly change the total probability of individual barrier if this is a real problem, and if, indeed, containment isolation failure is the proximate cause of cavitation and core melting, then we have a unit probability of containment failure as well.

I think that's the essence of the dilemma that we're in here. Now, --

MR. STUTZKE: Right. In my mind, I think it's the third bullet. It's the issue here of dependency, and when we think about the dependencies, one of the things that needs to be examined is the so-called balance between accident prevention and mitigation here because truly if you needed the overpressure credit realistically and the containment has failed and the scenario progresses to core damage, you have some type of a release, be it large or small or early or late, but you know the containment has failed.

And that's the dilemma here like this.

CHAIRMAN WALLIS: This is a LOCA, and you're main steamization valves are closed?

MR. STUTZKE: Yes. Well, that's one way to fail the containment.

CHAIRMAN WALLIS: But it could fail to close because of a piece of steam dryer that got in them? Did you consider that scenario?

MEMBER SIEBER: What's the probability of it?

CHAIRMAN WALLIS: I mean you could

construct. It's not an incredible event.

MR. LOBEL: There's two MSIVs on the BWRs. So you would have to destroy both the inside containment and the outside containment.

CHAIRMAN WALLIS: Yes. But you didn't consider this?

I think Entergy did consider MSIV closure, but I don't think they considered debris in it.

MR. STUTZKE: Right. I mean, I had modeled failure of MSIV closures as well, but not necessarily due to the debris.

CHAIRMAN WALLIS: And there's no chance the operator is is going to open it?

MR. STUTZKE: Possibly. I mean, I've looked at the physical construction of the MSIVs in the context of another issue the staff is pursuing, and it doesn't seem credible. The seat is up.

MEMBER DENNING: Continue.

MEMBER SIEBER: You would have to plug both valves with debris.

MR. STUTZKE: That's correct.

MEMBER SIEBER: You would have to have two chunks flowing eight feet apart at the same velocity to accomplish that. That to me seems incredible.

MEMBER DENNING: Go ahead, Marty.

MR. STUTZKE: Okay. Let's jump to Slide 10 now.

So I will try to go down these objectives briefly. There's no impact on any initiating event frequency or probability of preexisting containment leakage that would be created if the proposed credit is accepted like this because you haven't changed the normal operation of the power plant.

Similarly, if you use the baseline PRA, the so-called realistic assumptions, you don't need the credit. So you haven't changed the probability of failure of the fuel barrier or any other barrier. You haven't increased the risk, and you haven't changed the existing balance between prevention and mitigation.

The rub comes in, if you turn to the next slide, Rick, if you believe the alternative set of success criteria where the overpressure credit is really needed. Okay? You have to realize you're talking about at least four failures in order to get into core damage accident, the LOCA followed by failure of the containment integrity, failure of the suppression pool cooling, failure of the alternative injection sources. Okay?

CHAIRMAN WALLIS: This failure of

suppression pool cooling is apparently more than just one RHR train realistically.

MR. STUTZKE: That's right.

CHAIRMAN WALLIS: So it's failure of more than one RHR train.

MR. STUTZKE: That's right. When I say "pooling," I'm talking about the entire system. So there's multiple pumps.

CHAIRMAN WALLIS: To fail the suppression pool cooling, you have to fail two independent systems.

MR. STUTZKE: That's correct.

CHAIRMAN WALLIS: So you've really got five things here maybe.

MR. STUTZKE: But it's one of the uses of PRA. You see this by looking at minimal cut sets, and you look at the number of events in the cut set, and it takes a lot to get there.

The other thing that we've said before is the change, even if we assume the credit is necessary, the change in core damage frequency is small. The results appear to be robust. I've certainly looked at the uncertainties.

By the way, the numbers we're reporting here are mean values of parametric uncertainty distributions. They're not point estimates like this.

I presented to the subcommittee earlier it's possible to calculate the change in conditional containment failure probability, and again, it seems to be very small based on this.

MEMBER DENNING: Now, I'm going to interrupt you, Marty.

I think that we may have time to get to your conclusions, but indeed, it's pretty obvious. A good presentation.

MR. STUTZKE: Right.

MEMBER DENNING: Thank you.

I think we definitely want to go on and hear the next presentations. Shall we go ahead and do that now?

MR. LOBEL: How much time will I have?

MEMBER DENNING: We have until quarter of.

MR. LOBEL: Quarter of? Okay. I think I can get through.

Good afternoon. My name is Richard Lobel. I'm a senior reactor systems engineer in the Containment and Ventilation Branch in NRR.

Let me skip the purpose. I think we all know why we're here.

I want to go over the conclusion first,

and then I'll try to present the information that supports them. I'll show you that the crediting of containment accident pressure and calculating available net positive suction head for the Vermont Yankee extended power uprate arises from the conservative nature of the calculations that were done by the licensee, and that a more realistic but still conservative calculation would show the credit for containment --

CHAIRMAN WALLIS: Would you take out the "would" please and say "shows." I mean, do you have such a real calculation? Does it show or is it "would show" if it were performed? Is this a conditional sentence or what?

MR. LOBEL: I don't have --

CHAIRMAN WALLIS: I'm really bothered by that "would" in there.

MR. LOBEL: I have a --

CHAIRMAN WALLIS: -- statement.

MR. LOBEL: I have a calculation that's close to best estimate.

CHAIRMAN WALLIS: So you don't really know if it's true, do you?

I'm really worried about it. I think this is a very key argument. If it is true, I think that would influence me very much, but when you say "would show," I don't know if this calculation exists or not.

MR. LOBEL: That calculation that I have is -- well, it's more realistic, but still has some conservatism in it, and it shows even with the conservatism that containment pressure is not necessary.

CHAIRMAN WALLIS: But it still has some conservatism.

MR. LOBEL: It still has some conservatism.

CHAIRMAN WALLIS: Some. So it's no longer a bounding calculation.

MR. LOBEL: It's not a bounding calculation, right.

CHAIRMAN WALLIS: So we don't know --

MR. LOBEL: And I have a curve --

CHAIRMAN WALLIS: We don't know what the probability is of it being wrong, right?

MR. LOBEL: And I have a curve comparison with the bounding calculation.

CHAIRMAN WALLIS: So this is a vague statement. I thought it was a hard, really impressive statement, but I guess it's a little vaguer than that because we don't really know how uncertain giving up some conservative assumptions makes the --

MEMBER DENNING: When we see your results, we'll come back to this. I think you can move on and we'll come back.

MR. LOBEL: Okay.

CHAIRMAN WALLIS: But I'm just trying to get my rationale for doing A or B, you know, and if I really believe this statement, it makes a big difference to me.

MEMBER DENNING: I understand.

MR. LOBEL: Okay. Furthermore, a hypothetical single failure which results in loss of containment's capability to maintain accident pressure will not result in loss of NPSH margin, and I'll talk more about this later.

Credit for containment accident pressure has no impact on the operators since NPSH guidance in the Vermont Yankee emergency operating procedures already takes into account containment accident pressure, and so, therefore, based on conservative calculations done with acceptable analytic methods, the data and expert judgment of the ECCS pump vendor, consistency with the emergency operating procedures, and an acceptable level of risk, the staff finds that the licensee's proposal to credit containment accident pressure is acceptable.

Okay. The only point I wanted to make on this slide about Reg. Guide 1.82 I've already made. Keep in mind that what we're talking about here is just Vermont Yankee, and the conclusions apply just to Vermont Yankee, and we're not talking about the more general case where some of these statements may not hold.

And we're scheduled to come back to you again and talk about our revisions to the reg. guide early next year.

I've made this statement before on the next slide about regulations, that there is no regulation prohibiting credit for containment accident pressure for available NPS --

CHAIRMAN WALLIS: There's a whole reg. guide which does say that, isn't there, which has never been withdrawn?

MR. LOBEL: Well --

CHAIRMAN WALLIS: Yes. I know it's a reg. guide, but there is an old reg. guide.

MR. LOBEL: Yeah, there is an old reg. guide, safety guide, and as part of what we're trying to do with reg. guide 1.82, we're withdrawing -- well, not withdrawing -- we're going to put a note in Reg.

Guide 1.1, the old reg. guide, that will say that it shouldn't be used in the future. We're not withdrawing it because there are some licensees out there who still reference that reg. guide as part of their licensing basis.

Okay. Another point that I've made before that I'd like to restate is that boiling water reactor design basis accidents already credit containment integrity and containment accident pressure for other considerations. Radiological dose, analyses assume that the containment leaks at a rate L sub A that's defined in the regulations in Appendix J and in the tech. specs.

And Appendix K to Part 50 talks about minimizing containment pressure, not eliminating it, not assuming it isn't there, just minimizing it for the effectiveness of core spray cooling.

Okay. Now we're getting into more of the discussion that we've been talking about. This next slide is one example of the conservative nature of the calculation.

The licensee calculated the effect of considering the worst single failure. This was determined to be failure of an RHR heat exchanger outlet valve to open, which eliminates that heat exchanger.

So there are two trains of RHR. So that leaves one RHR heat exchanger for cooling the suppression pool. The resulting pressure pool temperature with all other variables at their limiting design basis values is 195 degrees Fahrenheit.

If instead we choose as the single failure the loss of the containment with all other variables at their limiting design basis values, then there are two RHR heat exchangers to cool the suppression pool. So the peak suppression pool temperature is 169 degrees Fahrenheit.

The licensee has determined that with a suppression pool temperature below 185 degrees Fahrenheit, credit for containment accident pressure is not needed. So with the worst single failure, the temperature of the suppression pool is 195 degrees, which is greater than 185 degrees. So containment pressure is needed for available NPSH with failure of the containment.

So assuming the containment is at atmospheric pressure with two trains of RHR now because I've already taken my single failure, the temperature I get is 169 degrees and credit for containment accident pressure isn't needed for NPSH.

Okay. Next slide.

I'm told this may be a unique way of looking at defense in depth, but because of the need to credit containment accident pressure for Vermont Yankee is due to the conservatism in the calculations, eliminating some of this conservatism would eliminate the need to credit pressure for NPSH margin.

And I just showed you the sample that changing the single failure from the worst single failure to the loss of containment pressure with all other conservative assumptions and input the same, adequate NPSH margin exists without crediting containment accident pressure.

So since the dependence between barriers is a function of the way the calculation is done and not a physical dependence, we consider that the defense in depth principle is maintained.

MEMBER POWERS: I want to ask you something on this. If you go through this analysis and, as I understand it, say you failed the containment, that reduces your sump pressure. You don't need the net positive suction head.

Do you get into a Part 100 problem?

MR. LOBEL: Well, you have to keep in mind -- sure, if you didn't have the containment, but you have to keep in mind, again, this is a design basis type analysis which is a stylized --

MEMBER POWERS: So was the Part 100 analysis.

MR. LOBEL: Right, right, but for each one you make a different set of assumptions that is limiting and sets the design of some parameters in the reactor or in the plant. So it's not surprising that there's an inconsistency from one analysis to another. Even maintaining containment pressure, for example, when you do the calculation for the peak containment pressure, you use a totally different set of assumptions, and the peak containment pressure -- I forgot the exact value for Vermont Yankee -- is around 43 psi. For the minimum pressure it's around 10 psi.

So I'm calculating the same parameter, but I'm interested in a different result. I'm interested in biasing my analysis to a different result, and so I get a far different analysis result.

That's not unusual in the way we do things.

MEMBER SIEBER: With the failure to cool containment and a failure of containment integrity, that's two failures which takes you beyond the design basis. Part 100 applies to --

MR. LOBEL: Was that the question? I missed the question.

MEMBER SIEBER: -- applies to the design basis.

MR. LOBEL: Yeah, right, and that's why I'm saying it's one failure or the other.

MEMBER SIEBER: Right.

MEMBER DENNING: I think you actually interpreted the question correctly.

MR. LOBEL: Okay. The next slide, the licensee provided the staff with some additional sensitivity studies to present to the committee. This first is related to the sensitivity I just discussed. It's a plot of the peak suppression pool temperature as a function of the service water temperature. The service water cools the RHR heat exchanger, which in turn cools the suppression pool.

The dotted horizontal line is the suppression pool temperature above which credit is needed for containment accident pressure for available NPSH, and this number, like I said, is 185 degrees.

Two other curves are plotted. The upper curve is the design basis peak suppression pool temperature as a function of the service water temperature. The assumed single failure is the failure of one RHR heat exchanger.

Notice that above a service water temperature of approximately 65 degrees, credit for containment accident pressure is necessary with this single failure.

The second curve is the same calculation, except that the assumed single failure is now loss of containment and, therefore, loss of containment accident pressure.

And notice that even if the maximum assumed service water temperature of 85 degrees, no credit for containment accident pressure is required since both RHR heat exchangers are available. So this is just another way of looking at what I presented on the previous slide.

MEMBER KRESS: How good do we know that 185 value?

MR. LOBEL: I'm sorry. What?

MEMBER KRESS: How good do we know the value of 185 as being the limit?

MR. LOBEL: I think I'll have to ask the licensee that question. It was their calculation. I can tell you though that the pre-extended power uprate temperature was 182.6 degrees and no containment pressure was needed. So it's close to another number that we know.

MEMBER KRESS: Yeah, that doesn't really answer my question.

MEMBER RANSOM: Just a point of clarification. The pink curve assumes both failures or only a single failure?

MEMBER KRESS: Single.

MR. LOBEL: They're different single failures.

MEMBER DENNING: We don't want to spend too much time on that containment single failure because it's kind of irrelevant, I think.

Did Entergy want to make any comments on the accuracy with which we know the 185, that that's the limit at which the NPSH requirement becomes an issue?

MR. NICHOLS: Craig Nichols from Entergy, Vermont Yankee.

I'd like to ask our lead on this, Mr. Bruce Slifer, to come up and address that question.

MR. SLIFER: Bruce Slifer from Vermont Yankee.

The temperature for the Archer pumps is based on the calculation of the available NPSH. So as temperature goes up, the available NPSH is reduced because of the increase in vapor pressure primarily. So what we did is an evaluation looking at the characteristics required of NPSH for both the core spray and the RHR pump.

The 185 degree limit is based upon actually the core spray pump being the most limiting pump for our case, and the calculated point at which you would lose available NPSH, assuming no credit for overpressure, would be 185 degrees.

MEMBER DENNING: But I think there are two conservatisms in there at least, one being the level of water in the suppression pool and the other being the temperature of water in the suppression pool. Is that true?

MR. SLIFER: Correct.

MEMBER DENNING: Inherent in 185?

MR. SLIFER: Well, there's several factors. It's the losses in the piping system, including the suction strainers and the debris on the strainers. This calculation was based on the maximum values for those, i.e., the highest calculated loss factors for all those conditions.

The suppression pool level was taken from the actual calculation of the containment response. So we assumed a certain value for that, and again, it was based upon the vendor's recommended values for the required NPSH.

MEMBER KRESS: That comes closer to answering my question because you calculate it based on pressure drops downstream that you get with a certain flow rate.

Now, I guess my question involves this. At 185 are you getting cavitation?

MR. SLIFER: Well, the limits are based upon the vendor recommended values, and at these operating conditions --

MEMBER KRESS: You will have some sort of flow reduction, but it will be enough --

MR. SLIFER: You will probably like their recommendations are based upon approximately a three percent head drop. So there is some head drop due to cavitation, but it's minimal.

MEMBER KRESS: But it's acceptable is what you're --

MR. SLIFER: It's acceptable. You can operate in these kinds of conditions for seven hours. After that they made a recommendation that the available NPSH should be higher.

MEMBER KRESS: And you've measured the pressure drop you get on those lines or is it calculated?

MR. SLIFER: This is calculated, supported by the periodic testing that we do, also compared to suction pressure at the pump inlet. Compared those against our calculations, we show that they're reasonable; the values are reasonable.

MEMBER KRESS: Do you actually test those sprays occasionally? This is the spray you're talking about.

MR. SLIFER: The core spray pumps and the arterial (phonetic) pumps are subjected to periodic testing on a quarterly basis.

MEMBER KRESS: And you measure flow and pressures during that?

MR. SLIFER: Yes, we do. We compare the flow requirements against a certain head requirement to assure that we're still operating within acceptable ranges.

MEMBER KRESS: Okay. Thank you.

MEMBER DENNING: Okay. Proceed. Thanks.

MR. LOBEL: Okay. The next slide is an illustration of the conservatism that goes into an input, and this historgram of the Vermont Yankee service water temperatures for the last approximately four years will illustrate that a little.

The histogram shows the percent of time that the service water temperature is at a specified value and also on the figure is a line representing a percentage of the time the service water is less than the given value, and notice that from the last slide the design basis calculation predicted that credit for containment accident pressure was needed when the service water temperature is greater than 65 degrees, and this is based on all the design conservative assumptions.

From this figure you can see that 69 percent of the time the service water temperature is less than 65 degrees. The design basis calculation uses a service water temperature of 85 degrees Fahrenheit. The service water temperature has never been at this value in the last four years. Ninety-eight percent of the time it has been below 80 degrees. Eighty-nine percent of the time it has been more than ten degrees below the value assumed in the design basis analysis.

Okay. Next. Next slide.

Okay. This next sensitivity study gets more to the realistic calculation. This sensitivity study shows the peak suppression pool temperature plotted against the service water temperature, again.

The single failure assumed is the design basis single failure of one RHR heat exchanger. The figure shows both the design basis calculation results, the solid line, and the results of a best estimate calculation, the dotted line.

And even though this is labeled the best estimate calculation, there's still some conservatism that's left that's still included.

The horizontal dotted line is the temperature above which credit for containment accident pressure is needed again, the 185 degrees.

At a service water temperature of 85 degrees, the assumed maximum value, the peak suppression pool temperature is 195 degrees, which is greater than 185 degrees, and so credit for containment accident pressure is needed.

For the best estimate calculation with a failure of one RHR heat exchanger, the peak suppression pool temperature doesn't reach 185 degrees until the service water is at its maximum assumed value.

So for the best estimate calculation, but assuming a single failure of one RHR heat exchanger, essentially no containment accident pressure is required.

VICE CHAIRMAN SHACK: Now, is this truly a best estimate or this is a best estimate 95 percent confidence?

MR. LOBEL: No, it's a best -- well, I don't know the confidence, but it's a best estimate, but still has some conservatisms. There's still a bounding feedwater energy addition that was left in. There's a cycle independent decay heat that was still left in. It assumes that five percent of the RHR heat exchanger tubes are plugged, and the bounding RHR fouling factor, and that the operators don't secure the ECCS pumps. So the pumps are operating, and they are adding their heat to the suppression pool also, which is significant.

So there's still some conservatism even in the best estimate calculation. So although it's labeled best estimate, it's still a little conservative, which I think goes to prove the point even more that a real best estimate calculation would be an even lower line and wouldn't need containment pressure at all. It's --

CHAIRMAN WALLIS: Now, if you actually used the probability distribution in your previous slide and you used it for some of the other inputs into this calculation, you could come up with a probabilistic output.

MR. LOBEL: Right.

CHAIRMAN WALLIS: And that would be a much more convincing argument. These sort of bits and pieces would show, well, if you take away this, it looks better. We haven't really got something that would show us how good it gets in reality.

MR. LOBEL: Well, yeah, that's right, and as you may recall when we were talking about Reg. Guide 1.82, that was one of the things that we added and we're hopeful that we're not going to be able to do something by February or March, but we're hopeful that some licensee will decide to try that approach or that --

CHAIRMAN WALLIS: Why don't we --

MR. LOBEL: -- try that ourselves.

CHAIRMAN WALLIS: Why don't we ask Vermont Yankee to do it? Do the full job?

MR. LOBEL: Well, actually I talked to Vermont Yankee not in terms of them doing it, but in terms of the idea of doing it about a year ago, and I can't speak for Vermont Yankee, but I think if we'd have all realized that the review was going to go on for this much more time that might have been a more feasible thing to try, and we hopefully could have gotten away from this whole controversy.

MEMBER DENNING: Let me ask a question, including the committee, and that is if you look at those things that are potentially variable, such as the suppression pool temperature, you know, normally we take a limiting value even for things like the initial suppression pool temperature.

MR. LOBEL: And they use a limiting --

MEMBER DENNING: And they use that there, and if you looked at variability over a year, that's a marked difference. I mean that in itself would bring down those temperatures with some high degree of probability by maybe ten or 15 degrees.

MR. LOBEL: My understanding is the level is controlled pretty carefully.

MEMBER DENNING: No, I meant -- did I say level? I meant the temperature.

MR. LOBEL: The temperature. I had --

MEMBER DENNING: The temperature prior to the event.

MR. LOBEL: Vermont Yankee gave me a curve of the temperature over time just like the service water one that we can provide to the committee.

CHAIRMAN WALLIS: And you get some benefit just like this one.

MEMBER DENNING: Yes.

CHAIRMAN WALLIS: So why don't you show them all together? Why do we get these bits and pieces if you only show us this piece.

MEMBER DENNING: Part of it is the question of how do you do a realistic estimate with uncertainties. Do you take things like you start at --

CHAIRMAN WALLIS: Do 59 runs.

MEMBER DENNING: Well, no. The question is do you say, okay, I'm going to start at the maximum possible suppression pool temperature, or do you say I'm going to look over your average and see realistically how does it vary, and include that in the probability.

And if you include that in the probability, it dramatically decreases the probability of exceeding it, but there still is kind of this regulatory inconsistency or I don't know. Perhaps people have really said this is how you do a realistic estimate with uncertainties.

MR. LOBEL: It's been done for other cases. It hasn't been done for this, but, for example, for calculating departure from nuclear boiling rations in PWRs, it's standard procedure now to do a calculation that's best estimate and then estimate the uncertainties in clad thickness and diameter and flow and pressure drop and things like that and then do just what we're talking about, add then all together at a 95-95 limit.

CHAIRMAN WALLIS: So why don't you require that they do it here? It can be done. It's just a question of another few weeks to do it or something, isn't it?

MR. LOBEL: Well, I think there's more to it than that, and you have to realize, too, that --

CHAIRMAN WALLIS: The computer program is there. Excuse me. they've obviously done a lot of calculations in sensitivity. So doing enough to do a full uncertainty analysis is just a matter of time. It's not a matter of something new.

MR. LOBEL: You have to realize, too that the purpose of doing a design basis analysis is to show that I've piled so much conservatism on that there's just no worry about whatever the bad outcome is.

So in those cases, licensees tend to pick bounding values where they can, and it may take a lot more effort to define a realistic value and an uncertainty.

A good example of that I would think of is debris blockage and pressure drop where experiments are done and analyses are done in a way to bound things. For example, Vermont Yankee in their analysis assumes that they lose all of the debris on half of the reactor coolant system. That's not a realistic assumption. It's a bounding assumption so that somebody doesn't have to look at every possible place where a pipe can break and calculate how much debris can come off from that break.

MEMBER DENNING: Why don't you do your summary slide and then we'll see if the committee has other inquiries

MR. LOBEL: Okay. Okay. The summary I already went through at the beginning, but in summary, based on a few considerations, the conservative calculations, acceptable analytic methods, the data and expert judgment of the ECCS pump vendor, consistency with emergency operating procedures, and an acceptable level of risk were the bases for the staff finding that the Vermont Yankee --

CHAIRMAN WALLIS: But you cannot have a conclusion based on something which would show if it were done. You cannot have a conclusion based on a "would show" argument. You've got to say it does show.

MR. LOBEL: Well --

CHAIRMAN WALLIS: And if it doesn't show, then it's not an argument.

MR. LOBEL: What I was trying to show with the slides that I had was that this need for containment pressure is really a figment of the way the calculation was done.

CHAIRMAN WALLIS: I think you're right.

MR. LOBEL: But what I was trying to say here is these are the reasons that we found that the licensee's use of --

CHAIRMAN WALLIS: Well, you see the problem I have is you're asking me to make a judgment that if you sort of did a little bit more of this, it would just get more conservative and everything would be even better.

But you're asking me to make judgment decisions when a little bit more effort would make me certain that I'm making the right decision.

MEMBER DENNING: I think the problem with your second sub-bullet is the way you're worded that you could say a more realistic but conservative calculation shows that credit is not needed.

But what you haven't taken into account is a probability. I mean, you've shown from your conservative calculation that you don't -- by removing conservatisms, I'm sorry, that you don't need it, but you haven't demonstrated it with a degree of confidence.

CHAIRMAN WALLIS: Well, I don't think he has because he's given up some conservatisms to do this realistic calculation. So it's not still conservative. only some things are still conservative.

So the whole calculation is not --

MR. LOBEL: Well, if you look at the slides that I presented to the Thermal Hydraulics Subcommittee when we were talking about the reg. guide I had something like eight pages of conservative --

CHAIRMAN WALLIS: I saw that.

MR. LOBEL: I'm sure you did.

-- of conservative assumptions, and here we're only talking about eliminating one at a time, and we still get the result that the analysis turns out to be that the need for overpressure is a figment of the analysis. Removing more conservatisms would --

CHAIRMAN WALLIS: So no one is ever going to do this full calculation which really wraps it up instead of having these bits and pieces which lead us to conclude that probably everything is okay?

MR. LOBEL: I don't have a realistic calculation in that sense. I don't think the licensee does, but they can answer for themselves.

CHAIRMAN WALLIS: I'm sorry, but in this new 182, you're going to ask for a realistic calculation with uncertainties, aren't you? A full realistic calculation with uncertainties, which you have not really got in this case. You have almost got it. It's within sight, but it's not quite there.

MR. LOBEL: I was hoping that this would be convincing enough that if you made that extra step, if taking away one conservatism did the job, then taking away a lot of conservatisms would be even better.

CHAIRMAN WALLIS: Well, I agree with that. That's a true statement.

MEMBER DENNING: And we do have information that was submitted to the subcommittee that has more examples of the magnitude of effective individual conservatisms.

MR. LOBEL: Part of the purpose for showing this was one of the criticisms from the subcommittee when I was showing those conservatisms was that I wasn't telling you how much each one was weighted, how much each one was worth. So part of the purpose of doing this was to show --

CHAIRMAN WALLIS: But you see my problem is when you take away a conservatism, unless you put an uncertainty on your new realism, you have given up something which no longer gives you a full argument, which we don't know how realistic the realistic estimate is. It may have a lot of uncertainty associated with it, in which case it's not as valuable as one which is more tightly understood. So just saying you've gone from conservative to realistic doesn't tell me very much until you put in the uncertainties in a logical way.

MEMBER DENNING: Are you ready now?

CHAIRMAN WALLIS: I'm sorry, but you know what I'm saying.

MEMBER DENNING: Now, if the committee agrees, we'll move now to the public comments.

Mr. Sherman, will you come and make a presentation to us?

MR. SHERMAN: Good afternoon. I'm Bill Sherman. I'm the state nuclear engineer for the State of Vermont, and with me today is Sara Huffman. She's the Director of Public Advocacy for the State of Vermont, and on behalf of the Douglas administration in Vermont, we appreciate your consideration of the issue of overpressure.

We from the beginning of the Vermont Yankee's application have been concerned about overpressure. We appreciate greatly the further analysis that the licensee has done in response to RAIs, appreciate greatly the work of the staff in looking at this, and your deliberation as well, and we will also stay with you for the rest of this week and weekend to see your deliberations and see how they play out.

I'll try and be as brief as I can with my nine slides here.

On the generic issue, the committee wrote a letter September 20th, 2005. I won't summarize the letter, but if you evaluate Vermont Yankee's proposal in accordance with the September letter, it doesn't appear to us that the proposal meets that letter. It appears to us that Vermont Yankee is asking for overpressure credit for longer than a few hours, that there are practical alternatives to being the overpressure credit, that there is not a full positive indication of containment integrity, and containment integrity has not been demonstrated for the credited time period.

And here's the curve that Vermont Yankee has put forth which shows that they're considering overpressure credit for a period of about 56 hours.

The staff response to the letter we're all aware of. Dr. Sharon came in in October, proposed a risk informed process for this. The State of Vermont believes that that has promise. As we stated in the power uprate subcommittee, we suspect that Entergy and the staff haven't analyzed the whole problem.

We talked about that at length at the subcommittee. We provided this chart which is modified. Actually number two is modified from the chart that we provided.

What we feel is that the new top event that should be reviewed should be pump fails due to inadequate NPSH. We feel that two cases for this top event should be evaluated, one case with overpressure credit, one case assuming that the practical alternative is implemented, that is, no overpressure credit, and we went through at the subcommittee, and I won't go through again how there's an uncertainty that is in each one of these items that, though we don't know what those uncertainties should be, we know that there is an uncertainty in each one of those items that could be considered, and we're not sure.

We haven't seen the staff's evaluation. So perhaps Mr. Stutzke has done all of this. He didn't respond to all of these items in his presentation today.

But let me just give one example of the pendency of our concern. What I've shown on this slide, which is too dense for you to read but each of you have in your own packets full size copies of this license event report. This is something that just came across our attention this week. This is a license event report for a three-quarter inch containment isolation valve which had been mispositioned open for ten years, nine years, I think, actually rather than ten years.

If you take time later and read the LER, you'll see that there are compensating measures why the fact that this was open may not have resulted in a problem, but it also shows you that in the real world things happen that are contrary to the overall plan.

This was a three-quarter inch valve. In my subcommittee presentation, I pointed out that the licensee had shown that a half inch valve, half inch opening in containment was what they calculated to defeat containment overpressure.

CHAIRMAN WALLIS: This was at Vermont

Yankee, this event?

MR. SHERMAN: This is correct.

CHAIRMAN WALLIS: They claimed they could detect leaks in containment.

MR. SHERMAN: This wasn't a leak. This was one of two valves that was mispositioned open, one of the two relied on containment isolation valves.

CHAIRMAN WALLIS: So it was making the containment not completely tight, was it?

PARTICIPANTS: No.

CHAIRMAN WALLIS: Oh, no? It was in series with another valve?

MR. SHERMAN: No, but it would feed into the probability of the containment not having integrity. It didn't defeat containment integrity, but it would feed into the probability.

CHAIRMAN WALLIS: It was on a different system?

MR. SHERMAN: It actually was on the RHR system that would be directly in play. It would have meant that in the LOCA situation that is under consideration you would have had only single valve protection. However, they're compensating additional valves downstream.

But I pointed out that if you're doing a risk informed evaluation, here's an example of something that feeds into that.

Now I'm going to go into something that I had planned. I didn't know of Mr. Stutzke's and Lobel's presentation, and unfortunately I'm going to be a little bit critical about what they said based on the next two slides.

The ATWS NPSH evaluation deserves a few more questions, I think. Most of what has been focused on is the LOCA NPSH evaluation. Let me just go to the next curve. This is the curve that the licensee provided for the ATWS, and let's just take a minute with it.

You can see at the bottom I put a time scale on the bottom. They need overpressure credit starting at 15 minutes, and they don't need it again after an hour and 15 minutes. It's an ATWS. They have 12 pounds pressure that they show.

Let me just flip back for a minute. On the LOCA curve they only showed eight pounds, seven and change of pressure available.

So now flipping back to the ATWS curve, you ask yourself a question at ATWS. ATWS has as much energy in it as a LOCA, and the way that ATWS develops pressure is a little bit different than a LOCA, but because of the way that it blows down through the relief valves, but you say to yourself that if the LOCA requires overpressure credit for 56 hours, then why does ATWS only require it for an hour and 15 minutes.

And the answer is because these are nominal values. These are not conservative values, and so what that means is that in Mr. Stutzke's presentation, using realistic assumptions to estimate, evaluate available NPSH, no containment overpressure credit is necessary. I don't believe that's true because I believe that these are realistic assumptions. I believe the nominal assumptions in ATWS show that overpressure is available.

Mr. Lobel said --

CHAIRMAN WALLIS: Is required.

MR. SHERMAN: Required. I'm sorry.

Mr. Lobel said that because the need to credit containment accident pressure for NPSH arises from the conservatisms in calculation, eliminating excess conservatisms eliminates the need to credit containment accident pressure, but I don't think that's right because I think that looking at the ATWS analysis, they need overpressure credit because this is a nominal analysis or realistic, if you like.

And my only point in showing this is this, that I'm not sure that the ATWS analysis meets the proposed Reg. Guide 182 change that the committee didn't accept because that proposed Reg. Guide 182 suggested that for overpressure they should do conservative calculations, maximize the temperature, minimize the pressure, but with ATWS they haven't done it, and if ATWS was done that way, you don't really know where it's going to come out compared to the LOCA.

And it tells us, the state, that we suspect that the best way to look at this is through the risk informed methodology that Dr. Sheeron (phonetic) suggests.

However, we suggest that the full evaluation of that, as we showed in this earlier slide, would be the better way to do it, taking into account some probability that the operator fails to retain, taking into account the probability that the debris head loss is more than expected, and maybe Mr. Stutzke's analysis did that. I don't think so, but maybe it did.

And I don't think that we know what the change in CDF would be. It might be in the ten to the minus eighth or ten to the minus seventh region. If they took in seismic, the seismic portion of that, it might not. It might be more in the ten to the minus fifth or sixth area, and therefore questionable in whether it was desirable.

Here's my summary. Under the ACRS letter that you wrote, we don't think overpressure should be granted. Under Dr. Sheeron's proposal, we still are troubled by the modification of defense in depth.

The answer to the question that somebody asked a minute ago about is Appendix 1 or is 10 CFR 100 affected, well, 10 CFR 100 is affected if you fail containment and you needed overpressure credit. Then 10 CFR 100 is affected because you're apt to have those two failures result in fuel failure.

Ten CFR 100 is not affected if you fail containment, but your pumps, your ECCS pumps don't depend on overpressure.

At any rate, Item No. 2, if the whole problem were analyzed, we'd think that we'd have more light on the problem.

MEMBER DENNING: Thank you.

MR. SHERMAN: thank you.

MEMBER DENNING: Bill, we'd like to thank you for your thoughtful input throughout this process. Thank you.

MR. SHERMAN: Thank you very much.

MEMBER DENNING: Mr. Shadis, are you available?

And, again, I'll ask you to be brief, although I realize that you do have some important things to present to us.

CHAIRMAN WALLIS: Well, how do we handle something, Mr. Chairman of this session, when the new question is raised, say, about ATWS? Can we ask the staff to respond to that? I don't know --

MEMBER DENNING: We certainly can --

CHAIRMAN WALLIS: He's raised a new question here.

MEMBER DENNING: Well, actually not a new question on ATWS.

CHAIRMAN WALLIS: He said the credit is needed even with a realistic -- you know, which is not what they were claiming. So are we going to hear from the staff on that?

MEMBER DENNING: Well, we're going to have to discuss that.

CHAIRMAN WALLIS: Or are we going to make that decision ourselves?

MEMBER SIEBER: We'll discuss it.

MEMBER DENNING: Mr. Shadis, will you

proceed?

MR. SHADIS: Thank you.

As a preliminary, just to pick up on one comment I heard in the earlier discussion with respect to debris from a failed skin dryer impacting the ability of the MSIVs to close, and one panel member suggested that having two pieces arrive eight foot apart at the same speed and the same time was not a credible event, I just want to remind you that the first catastrophic failure of the steam dryer at Quad Cities, a piece of steel nine feet in length and 18 inches in diameter was shed, and that folding up on the outboard MSIV, number one, could affect two MSIVs, but secondly, could form a trap for following debris.

I think the image that these pieces would be small and discrete may be nonconservative.

My topic, again, and I spoke to the subcommittee on this, is the question of the NRC's pilot program inspection that was conducted at Vermont Yankee, and this inspection program done in August of 2004, according to the SECY paper issued July 1st was done in support of the uprate review, and items were selected particularly to support uprate review.

The conclusion of the NRC staff conducting that inspection was that, and their opinion, too much reliance was placed on representations from the licensees that were not confirmed by actual physical inspection, and they noted, too, that I think with some degree of surprise that there seemed to be still latent design issues emerging at all of the power plants that were part of that pilot inspection.

This committee may know that the Vermont Public Service Board is anticipating that the committee will at some level review the engineering design pilot inspection that was done at Vermont Yankee and give some opinion of it.

That inspection was also completed in part to address a request from the Vermont Public Service Board for what they termed an independent engineering assessment, and that was a mini diagnostic evaluation team type of assessment where four systems were to be gone through in a deep vertical slice inspection.

They asked for it to be an independent assessment, and independence was there, but it consisted in that inspection of requiring that people who had contact within the previous two years with the licensee would be excluded from the inspection team, the licensee or the owner-operator Entergy.

And this is a step back from the kind of independence that was exhibited when the Maine Yankee independent safety assessment, which was also a diagnostic evaluation team derived inspection, was done. In that case there was no one permitted to be on the team from either Region 1 or the Office of Nuclear Reactor Regulation.

So I just offer that comment. I have provided for you a rough outline. They were intended to be viewgraphs, and we didn't get that far. However, I am hoping that this committee will, for the benefit of the Vermont Public Service Board and the people of Vermont, draw some kind of critique or evaluation of that inspection report.

And finally, I'm sorry to repeat, but it appears to be a matter of conviction at NRC still that the plants as they are represented in licensee documentation are the plants as they would be found in a physical inspection, and that not only goes to the physical components of the plant, but it also goes to the actions that are represented in the licensee's applications.

For example, at Vermont Yankee, one issue was the restoration of off-site power and how long it would take to switch over to an alternative power source. Another issue that arose was the question of how much time it would take to establish a remote control panel and set up to operate the reactor should the control room have to be abandoned because of radiological, well, habitability considerations.

And it proved to be upon actual inspection that what the licensee was relying on and representing in their application was not true, was not the case. So I guess the appeal here is that in reviewing this, this committee consider the recommendation that all extended power uprates be underwritten with a real diagnostic, physical, on-site examination.

Thank you. That concludes my remarks.

MEMBER DENNING: Thank you, and I'd also like to thank you, Mr. Shadis, for your input, particularly the experience that you have related to us that related to the Maine Yankee. Thank you very much.

MR. SHADIS: Thank you.

MEMBER POWERS: Dr. Denning, did the subcommittee look at the issue of unfiltered inlaid heat (phonetic) in the control room at Vermont Yankee?

MEMBER DENNING: I'm sorry. Did we look at?

MEMBER POWERS: Unfiltered inlaid heat. There are a lot of other control things are well off their design specs, and I just wondered where this client stood.

MEMBER DENNING: I don't know the answer. Can Entergy make a comment about that?

MR. PEREZ: Good afternoon. My name is Pedro Perez, representing Vermont Yankee.

At the Vermont Yankee plant there is no control room filtration, such as charcoal or HEPA filters. We assumed when we implemented the alternate source term that basically the control room is left open up to the full ventilation flow rate. So in principle everything is unfiltered that comes into the control room, and we meet the habitability requirements.

MEMBER KRESS: By using face masks?

MR. PEDRO: No, sir. No KI and no SCBAs.

MEMBER POWERS: You can do it with IST.

MR. PEDRO: With the IST.

MEMBER DENNING: Thank you very much.

MR. PEDRO: You're welcome.

MEMBER POWERS: Probably wrong.

MEMBER DENNING: Those are the only two comments that we had requested from the public. Does anyone else from the public have any comments?

(No response.)

MEMBER DENNING: Thank you, and I turn it back to you.

CHAIRMAN WALLIS: I would like to know when we'll hear an answer to this ATWS question, and apparently ATWS does require overpressure --

MEMBER DENNING: No.

CHAIRMAN WALLIS: If you take it away does it affect the CDF?

MEMBER DENNING: If you look at those things that reduce the suppression pool temperature associated with the large local, most of those things have applicability to the ATWS. If you want to reduce that --

CHAIRMAN WALLIS: Yes, but then we have a CDF calculation which gets to the borderline of some region if we add on the ATWS.

Did the staff consider this at all or are we --

MEMBER DENNING: Oh, yes. We had a presentation on ATWS, but it was not -- the focus was much more on the --

CHAIRMAN WALLIS: It wasn't focused on NPSH was it?

MEMBER DENNING: What's that?

CHAIRMAN WALLIS: It wasn't focused on the NPSH.

MEMBER DENNING: It wasn't, and there was presentation related to NPSH, but the focus was on the large LOCA just because it required more pressure for a much more extended period of time.

CHAIRMAN WALLIS: I know.

MEMBER DENNING: Did you want to --

CHAIRMAN WALLIS: I was wondering if the staff's conclusions that they presented to us are changed by the points that were made here about ATWS.

MEMBER SIEBER: I don't think you get into recirculation during an ATWS event because of the short time that there is pressure relief, and the minimum amount --

MEMBER DENNING: The staff will make a response on that.

MR. LOBEL: This is Richard Lobel of the staff.

We did look at the ATWS calculation. I haven't looked at it lately, but Mr. Sherman is correct that the ATWS is supposed to be or can be analyzed with nominal realistic values, but Vermont Yankee did use some conservative assumptions. They used the maximum flow rate for the pump. They considered that the debris from the LOCA was on the ECCS strainers even though the only debris that would be generated would be from the lifting of a safety valve, not from the breakage of the largest pipe.

Maybe they can help me. I'm not sure about these two, but I think the minimum suppression pool level was assumed and only one heat exchanger was assumed in the ATWS analysis.

MR. DREYFUSS: John Dreyfuss, Director of Engineering, VY.

We'd like to provide some insight on this question as well.

Craig.

MR. NICHOLS: Craig Nichols, Entergy, Vermont Yankee.

And we do have several folks here who were involved in that analysis, and Mr. Lobel is correct that obviously the ATWS is a beyond design basis event, which includes a single failure right off the bat of both the RPS primary and secondary system failure in describing the reactors. So we start from that position.

Our analysis did include similar to the LOCA analysis the design basis service water temperature, torus temperature and level; it shows a higher decay heat rate, et cetera. So there were many evaluations or parts of the evaluation that did include conservative values up to and including the tech. spec. value similar to the design basis LOCA.

We also did do a PSA of the ATWS, and we have people here that can also discuss the public safety assessment that was done for containment overpressure related to the ATWS.

So if the staff have particular questions or the ACRS committee has particular questions, we could assemble folks to discuss that.

CHAIRMAN WALLIS: What I was concerned about was these conclusions on your slide, which we might even quote in our letter not being true if you included ATWS. That's what concerned me, saying something which is not completely valid in our letter or relying on a statement from you which is no longer quite true as it was before.

MR. LOBEL: Well, I was aware of the ATWS situation. I was debating whether to put that in the presentation. I was trying to keep the presentation focused, and I don't believe that because of the conservatisms that we've just mentioned that if you took those conservatisms out that it would change the conclusion, you wouldn't need containment pressure.

If I would have thought differently, I would have mentioned it and I wouldn't have made such a point of those conclusions.

MEMBER SIEBER: Has the licensee asked for an exemption in the ATWS situation?

MR. LOBEL: An exemption for?

MEMBER SIEBER: For overpresssure, containment overpressure.

MR. LOBEL: You mean for crediting overpressure?

MEMBER SIEBER: Yes.

MR. LOBEL: Yes, but that's based on --

MEMBER SIEBER: For ATWS?

MR. LOBEL: For ATWS, but that was the curve that Mr. Sherman showed, but that was based on the analysis we're talking about that had these conservative assumptions in it.

VICE CHAIRMAN SHACK: Again, their Table 3.3 in their PRA analysis says that the ATWS contribution, if you credit or don't credit the overpressure, is 2.9 times ten to the minus ten.

MEMBER KRESS: The CDR.

MEMBER SIEBER: Pretty likely.

VICE CHAIRMAN SHACK: CDF.

MR. LOBEL: This is Richard Lobel again.

Let me say, too, that just so we're clear, I think I mentioned this at the subcommittee, but there were two other events, the Appendix R fire and the station blackout that the licensee originally credited containment overpressure and then revised their analyses by crediting another service water pump that changed that analysis.

MEMBER DENNING: Thank you for that full disclosure.

Okay. Thank you, Graham.

CHAIRMAN WALLIS: Okay. I don't think we have anything else we have to do at this time. I'd like to recess, and we are supposed to be back at four o'clock. We do not need the transcript from now on.

Thank you very much.

(Whereupon, at 3:22 p.m., the Advisory Committee meeting was adjourned.)

CERTIFICATE

This is to certify that the attached proceedings before the United States Nuclear Regulatory Commission in the matter of:

Name of Proceeding: Advisory Committee on Reactor Safeguards 528th Meeting

Docket Number: n/a

Location: Rockville, MD

were held as herein appears, and that this is the original transcript thereof for the file of the United States Nuclear Regulatory Commission taken by me and, thereafter reduced to typewriting by me or under the direction of the court reporting company, and that the transcript is a true and accurate record of the foregoing proceedings.

Tobias Walter
Official Reporter
Neal R. Gross & Co., Inc.

Entergy Vermont Yankee

Extended Power Uprate

Presentation to the

Advisory Committee on Reactor Safeguards

December 7, 2005

Entergy

Steam Dryer

Key Points

- Acoustic Loads are Primary Source of Dryer Significant Degradation
- VY Measurement Configuration Detects Acoustic Loads
- VY Dryer:
  - No CLTP Acoustic Resonance
  - Substantial Margin to ASME Stress Limit
  - Baseline Inspection – No Structural Vulnerabilities
  - Modified to Strengthen for EPU Operation
  - Power Ascension Controlled via Monitoring Plan

Steam Dryer

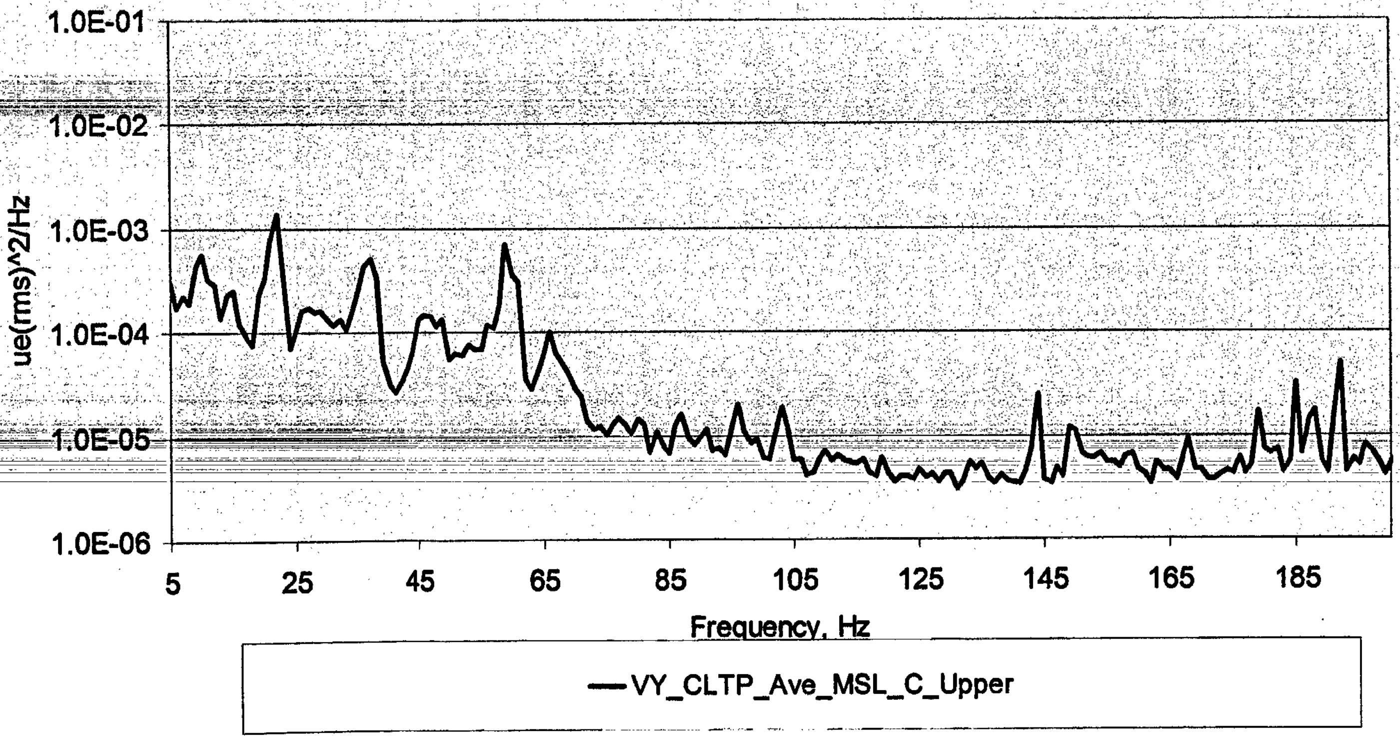
VY Main Steam Vibration

- MSL Measurements
  - Strain Gages
  - Accelerometers
- Main Steam Branch Line Potential Acoustic Resonators
- MSL Monitoring Will Detect Excitation from Sources in:
  - Main Steam Lines
  - Reactor Vessel

Steam Dryer

SG Measurements for Acoustic Monitoring

Figure 1a: VY SG Data CLTP

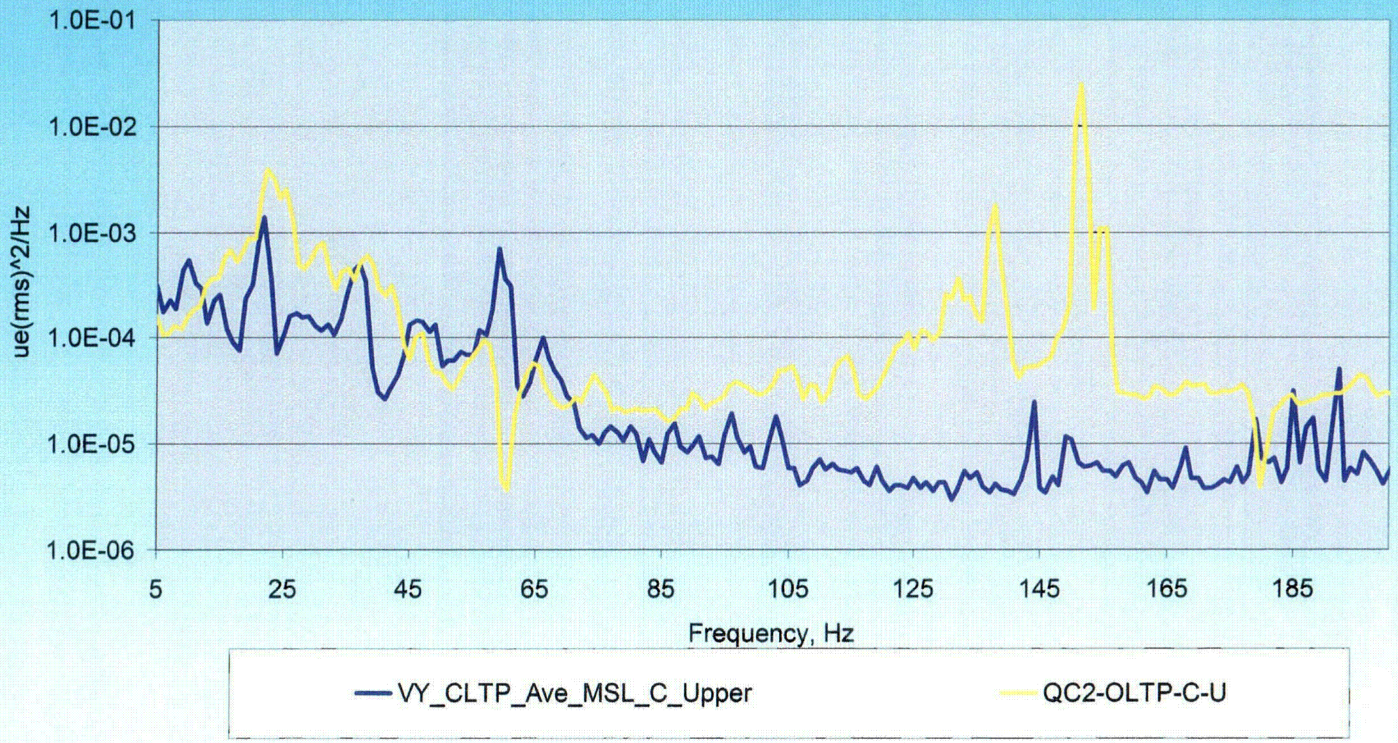


Entergy

Steam Dryer

SG Measurements for Acoustic Monitoring

Figure 1b: VY SG Data and QC OLTP



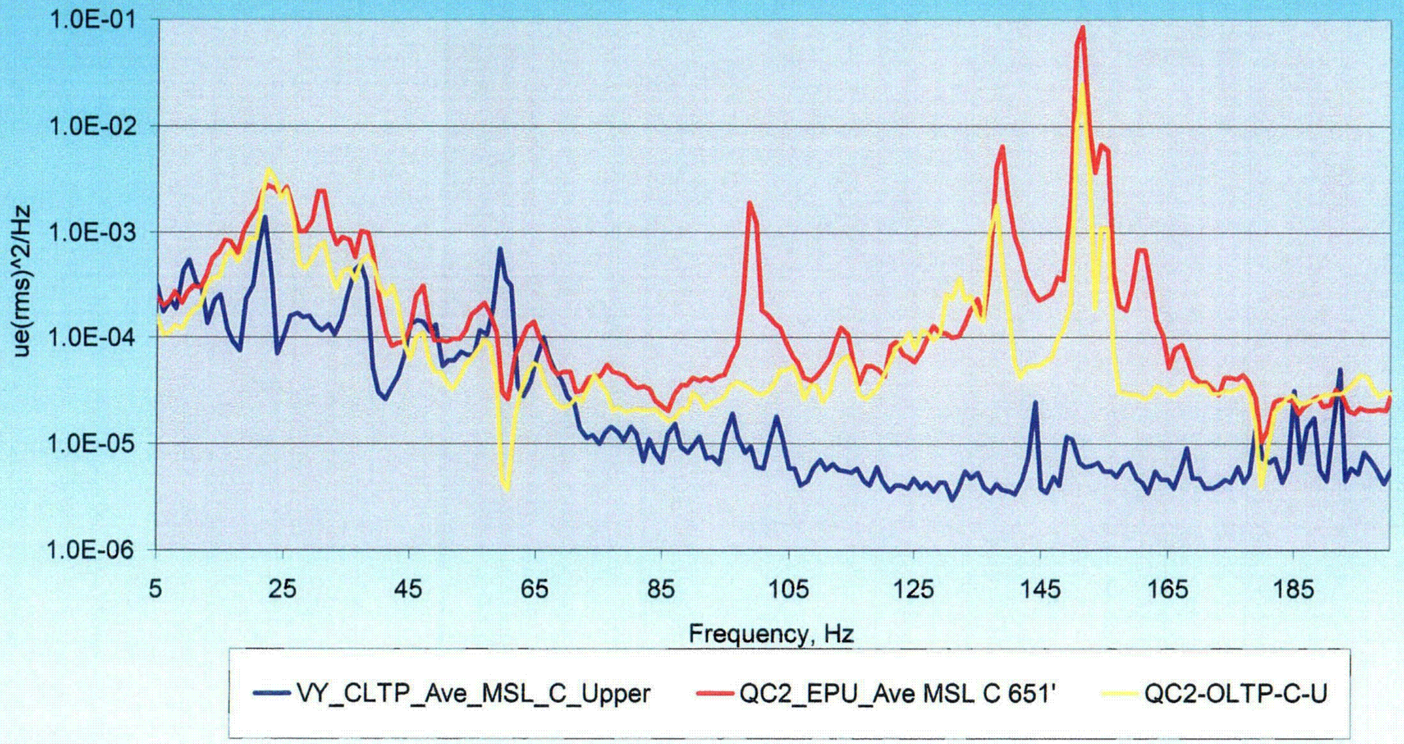
C01

Entergy

Steam Dryer

SG Measurements for Acoustic Monitoring

Figure 1c: VY SG Data and QC OLTP & EPU



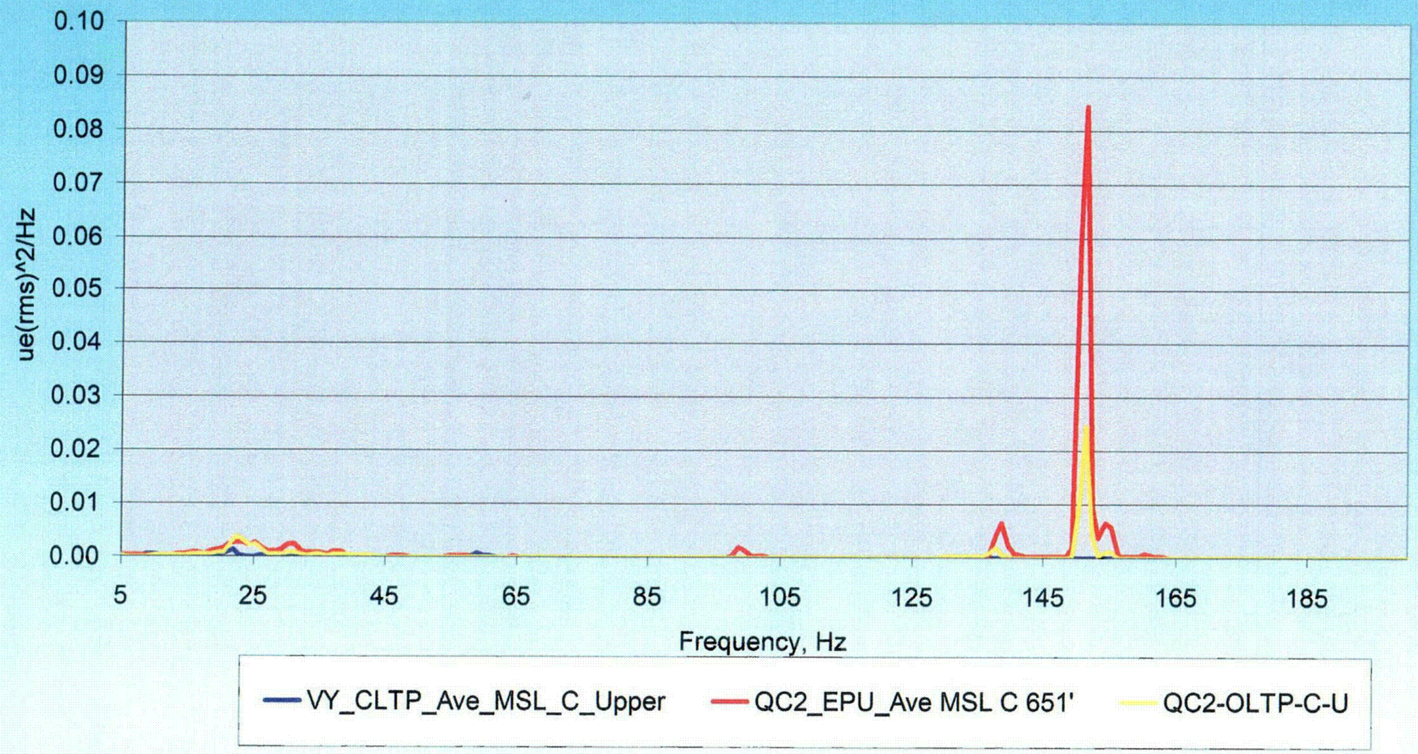
C02

Entergy

Steam Dryer

SG Measurements for Acoustic Monitoring

Figure 1d: VY SG Data and QC OLTP & EPU (linear)



C03

Steam Dryer

VY Structural Analysis – Load Definition

- Acoustic Loads Impact Vertical Faces of the Dryer
  - Generated from Steam Line Data
  - Transfer of Steam Line Data Benchmarked via QC Data

- Turbulent Forces Act in the Area of the Nozzle
  - Little Effect on Dryer Components

Steam Dryer

VY Structural Analysis – Peak Stress & LCF

VY Dryer Limiting Component – Vertical Face Top Weld

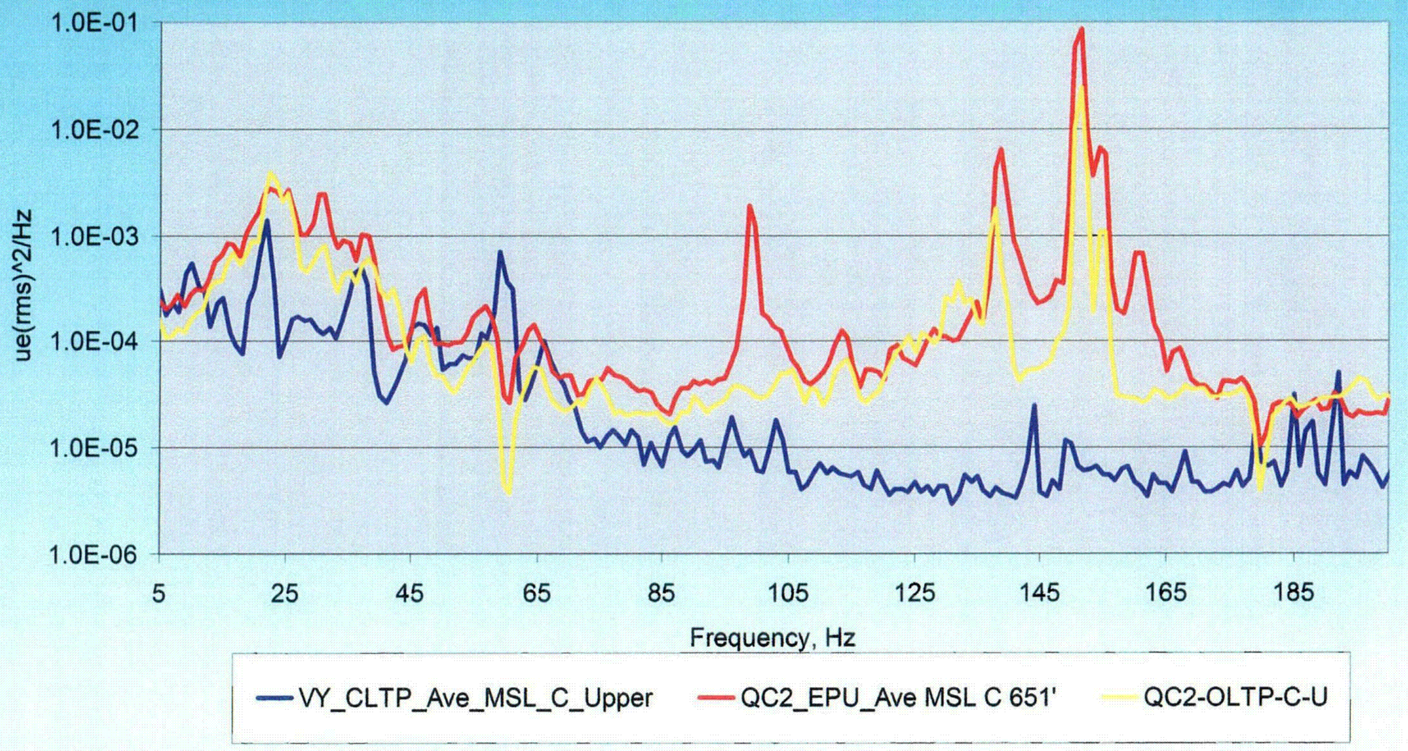
- Peak Calculated Stress 5,450 psi
- ASME Acceptance Criterion 13,600 psi
- Limit for Power Ascension 7,400 psi
  - Reflects 2.78 Limit Curve Factor

Entergy

Steam Dryer

SG Measurements & Power Ascension Limit Curve

Figure 2a: VY SG Data and QC OLTP & EPU



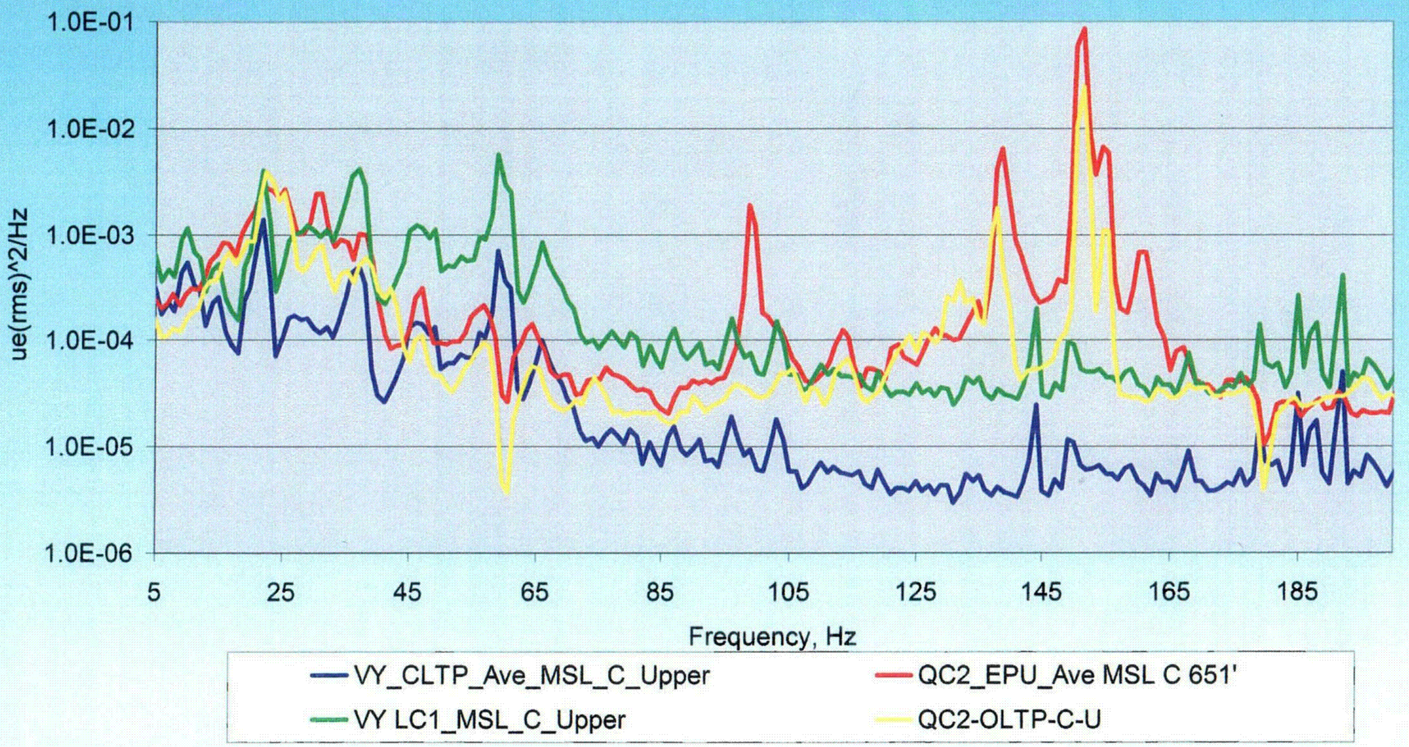
C04

Entergy

Steam Dryer

SG Measurements & Power Ascension Limit Curve

Figure 2b: VY SG Data and QC OLTP & EPU

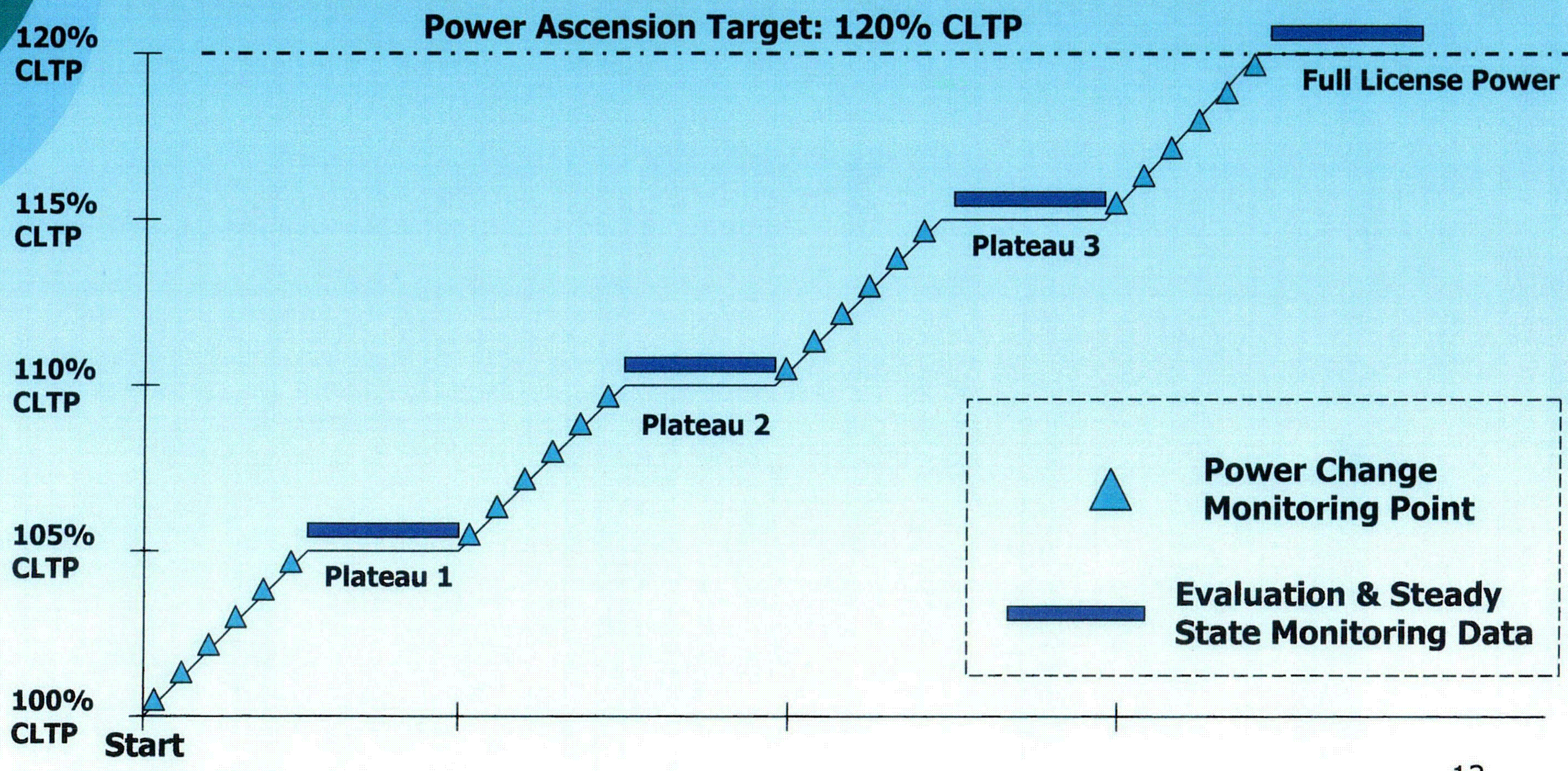


C05

Steam Dryer

Power Ascension Monitoring – Test Plateaus

Figure 3: Dryer Monitoring & Test Plateaus



c06

Steam Dryer

Conclusions

- No Vulnerability at CLTP
- VY Measures Acoustic Loads
- Margin to Fatigue Stress Limit

ACRS Full Committee Meeting

NRC Staff Review of Proposed Extended Power Uprate
For
Vermont Yankee Nuclear Power Station

December 7, 2005

Risk Evaluation of Proposed Credit for Containment Accident Pressure

Martin A. Stutzke
Senior Reliability & Risk Analyst
Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

In a Nutshell...

- Entergy has completed its risk evaluation of the proposed credit for containment accident pressure (CAP) to provide adequate net positive suction head (NPSH) to the emergency core cooling system (ECCS) pumps.
- Using realistic assumptions to estimate available NPSH, no CAP credit is necessary. Thus, granting the proposed CAP credit does not increase the risk associated with operation of Vermont Yankee (VY).
- The proposed CAP credit meets the five key principles of risk-informed decisionmaking.

Entergy's Risk Evaluation

- Chronology:

10/05/2005	Staff asks Entergy to provide risk evaluation that addresses the five key principles of risk-informed decisionmaking in RG 1.174
10/21/2005	Entergy provides partial risk evaluation (Supplement 38)
10/26/2005	Entergy completes risk evaluation (Supplement 39)
11/25/2005	Staff issues RAI about the risk evaluation
12/02/2005	Entergy responds to the staff RAI (Supplement 43)

- To address PRA modeling uncertainty introduced by the proposed CAP credit, Entergy performed a sensitivity analysis.

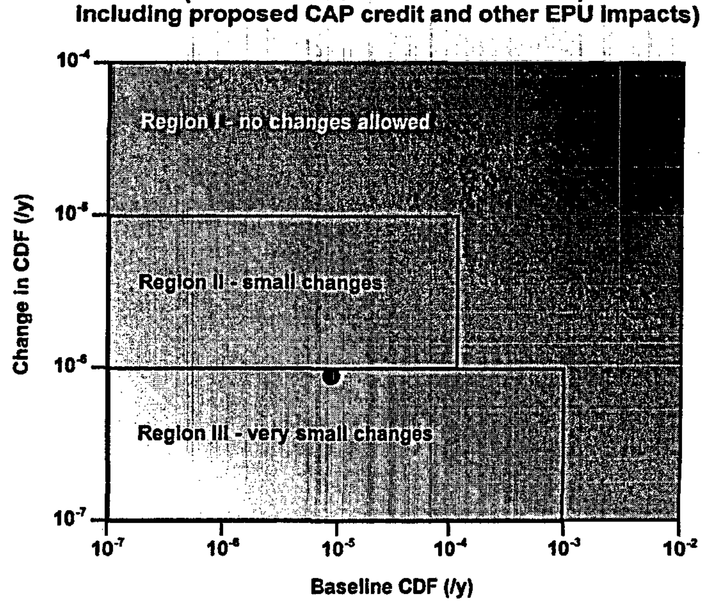
Entergy's Risk Evaluation (continued)

- Differences between the Entergy and staff sensitivity analyses:

	Entergy	Staff
Credit for alternative injection (AI) sources after loss of LPCI and CS pumps due to loss of containment integrity	yes	no
Note: No credit for AI after large LOCA for injection systems that take suction outside the suppression pool		
Credit for suppression pool cooling following loss of containment integrity	no	yes
Probability of pre-existing containment leakage		
Failure hole size	60 x La	35 x La
Failure probability	2.5E-04	1.4E-02
Data source	EPRI TR-1009325 December 2003	NEI Intermin Guidance 11/13/2001
Basis	expert elicitation	Bayesian (Jeffrey's non-information prior; no failures in 182 tests)
Change in CDF	5.8E-07	6.2E-08

Entergy's Risk Evaluation (continued)

Entergy's Sensitivity Analysis
(internal events and internal floods;
including proposed CAP credit and other EPU impacts)



Five Key Principles of Risk-Informed Decisionmaking in RG 1.174 and SRP 19

- The five key principles:
  - Proposed change meets the current regulations.
  - Proposed change is consistent with the defense-in-depth philosophy.
  - Proposed change maintains sufficient safety margins.
  - Increases in risk should be small and consistent with the intent of the Commission's Safety Goal Policy Statement (51 FR 30028).
  - Impact of proposed change should be monitored using performance measurement strategies.
- Acceptability of proposed change is determined by an integrated decisionmaking process.

Integrated Decisionmaking

- RG 1.174, Section 2.2.6 discusses integrated decisionmaking, and states that "None of the individual analyses is sufficient in and of itself."
- ACRS guidance:
  - ACRS letter of May 19, 1999 expressed concerns about the staff making "arbitrary appeals to defense in depth" to avoid making changes to regulations and regulatory practices that seemed appropriate in light of PRA results.
  - Joint ACNW/ACRS letter of May 25, 2000 discussed establishing limits of necessity and sufficiency on defense-in-depth within a risk-informed regulatory framework.

Defense-in-Depth Evaluation

- The proposed CAP credit is consistent with the defense-in-depth philosophy because it meets the four defense-in-depth objectives stated in SRP 19:
  - Does not result in a significant increase in the existing challenges to the integrity of barriers.
  - Does not significantly change the failure probability of any individual barrier.
  - Does not introduce new or additional failure dependencies among barriers that significantly increase the likelihood of failure compared to existing conditions.
  - Overall redundancy and diversity among barriers is sufficient to ensure compatibility with risk acceptance guidelines.

9

Defense-in-Depth Evaluation (continued)

- The proposed CAP credit does not affect normal plant operating conditions. So, no impact on:
  - Frequency of any initiating event, or
  - Probability of pre-existing containment leakage.
- Using realistic assumptions to estimate available NPSH, no CAP credit is necessary. Therefore, the proposed CAP credit does not:
  - Change the failure probability of the fuel barrier,
  - Increase the risk of VY operations, or
  - Significantly change the existing balance between accident prevention and mitigation.

10

Defense-in-Depth Evaluation (continued)

- Even if the CAP credit is assumed to change in PRA success criteria, then:
  - There must be at least four failures to cause a core-damage accident (LOCA followed by loss of containment integrity, suppression pool cooling, and alternative injection sources).
  - The change in CDF is very small and meets the RG 1.174 risk acceptance guidelines.
  - Results are robust in terms of uncertainties and sensitivities to key modeling parameters and assumptions.
  - No significant change in conditional containment failure probability (CCFP).

Performance Monitoring

- Diverse methods for detecting containment leakage:
  - Drywell/torus air space differential pressure > 1.7 psi - control room alarm; measured once per shift.
  - Low torus air space pressure - continuous.
  - Unusual nitrogen makeup - measured daily.
  - Oxygen concentration ≥ 4% - weekly measurements.
  - Integrated leak rate tests (ILRT) - Type A test done once every 15 years (temporary change).
- The fraction of time that the plant would be operated with a containment leak is small:
  - Leaks will be promptly detected, and
  - TS preclude prolonged operation with known leaks.
- Leak detection not explicitly considered in PRA.

Conclusions

- Entergy has completed its risk evaluation of the proposed credit for CAP to provide adequate NPSH to the ECCS pumps.
- Using realistic assumptions to estimate available NPSH, no CAP credit is necessary. Thus, granting the proposed CAP credit does not increase the risk associated with operation of VY.
- The proposed CAP credit meets the five key principles of risk-informed decisionmaking.

Deterministic Evaluation of Proposed Credit for Containment Accident Pressure

Richard Lobel
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Probabilistic Safety Assessment Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Purpose

- To discuss NRC staff review of Entergy's proposal to credit containment accident pressure in determining available net positive suction head (NPSH) for emergency core cooling system (ECCS) pumps for certain Vermont Yankee (VY) design basis accidents (DBAs).

Conclusions of NRC Review

- Need for crediting containment accident pressure for VY arises from conservative nature of design basis analyses.
- A more realistic, but still conservative, calculation would show that credit is not needed.
- A single failure resulting in loss of containment integrity will not result in a loss of NPSH margin.
- Credit for containment accident pressure has no impact on the operators.
- NRC staff finds proposed crediting of containment accident pressure for VY to be acceptable.

Regulatory Guide (RG) 1.82

- RG 1.82 is currently being revised by the NRC staff to address ACRS concerns.
- Entergy has stated as part of VY extended power uprate (EPU) submittals that it does not intend to make RG 1.82 part of the VY licensing basis.
- Methods and solutions different from those set out in RGs will be acceptable to the staff if they provide a basis for the requisite safety findings.
- Bottom line - all unresolved issues regarding revisions to RG 1.82 do not need to be resolved to find VY proposal acceptable.

Regulations

- There is no regulation prohibiting credit for containment accident pressure in determing available NPSH for safety-related pumps.

Design Basis Accidents

- Boiling Water Reactor (BWR) DBAs currently credit containment integrity and containment accident pressure for other considerations:
  - Radiological dose
  - Effectiveness of core spray cooling

Single Failure Considerations

Single Failure	Peak Suppression Pool Temp
RHR Heat Exchanger	195 F
Containment*	169 F

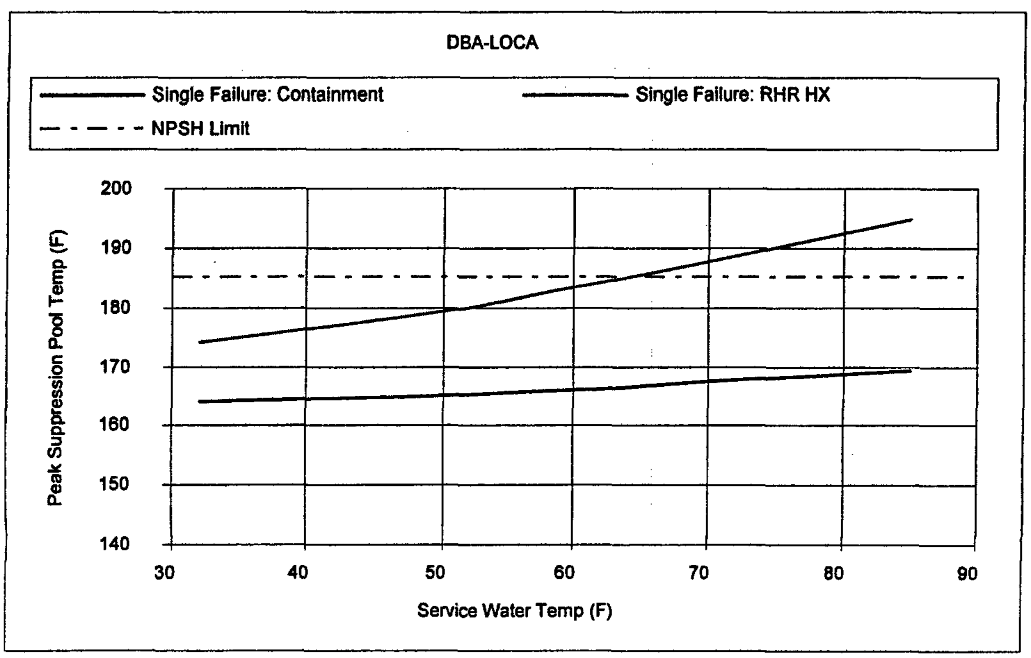
*If it is assumed that a single failure causes a loss of containment integrity, both RHR heat exchangers would be available and peak suppression pool temperature would be 169 F.

Credit for containment accident pressure is not needed if the suppression pool temperature is less than 185 F.

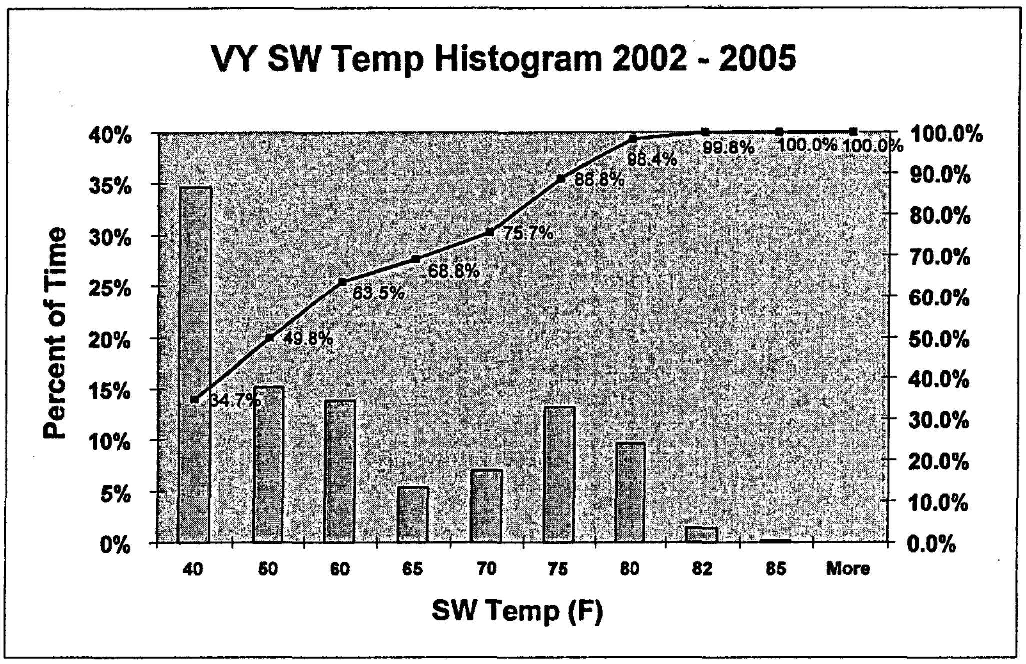
Defense-in-Depth

- Because the need to credit containment accident pressure for NPSH arises from the conservatisms in the calculations, eliminating excess conservatism eliminates the need to credit containment accident pressure.
- Dependence between barriers has been raised as an issue, however for VY based on the way calculation is done, there is no realistic physical dependence between barriers.
- Therefore, NRC staff considers defense-in-depth is maintained.

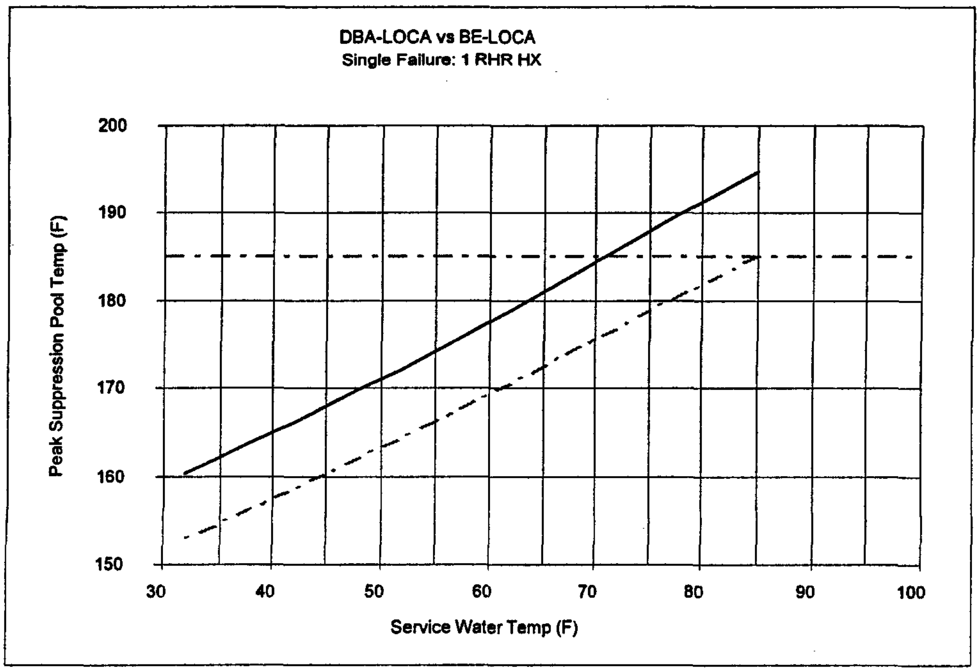
VY Sensitivity Study



VY Sensitivity Study (continued)



VY Sensitivity Study (continued)



Summary

- NRC staff finds that credit for containment accident pressure for VY is acceptable and is based on conservative calculations:
  - These calculations result in the need to credit containment accident pressure
  - A more realistic, but still conservative, calculation would show that credit is not needed
- Single failure resulting in a loss of containment integrity will not result in a loss of NPSH margin.

ACRS Full Committee

Vermont Yankee Extended Power Uprate Containment Overpressure Credit

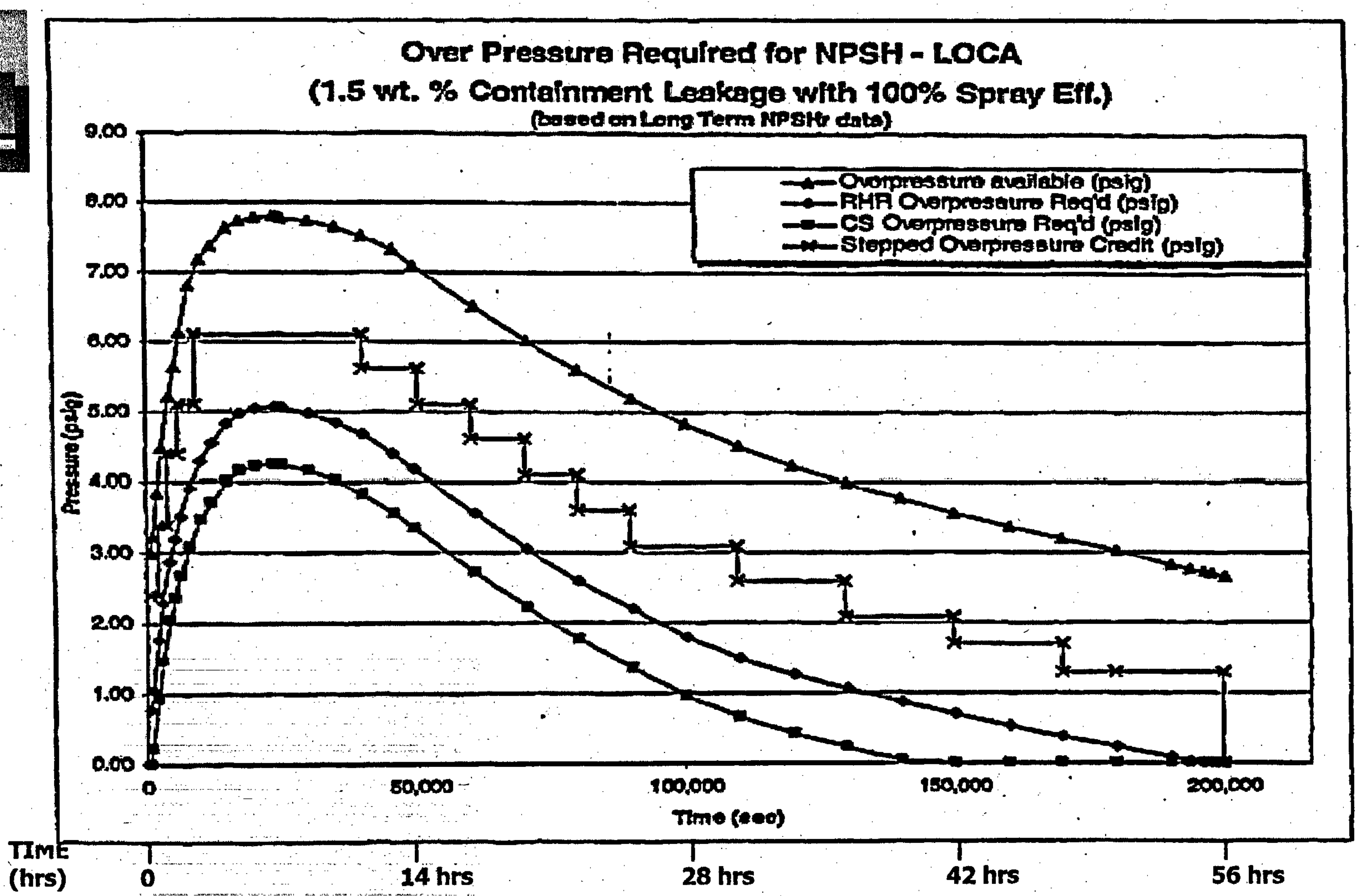
December 7, 2005

Bill Sherman – VT Dept of Public Service

Proposal Does Not Appear to Meet ACRS Letter September 20, 2005

- Longer than a few hours
- There are practical alternatives
- There is not full positive indication of containment integrity
- Containment integrity has not been demonstrated for the credited time period

Figure 4.2 LOCA – Long Term (1.5 wt. % Containment Leakage & 100% Spray Efficiency)



Staff Response to ACRS Letter

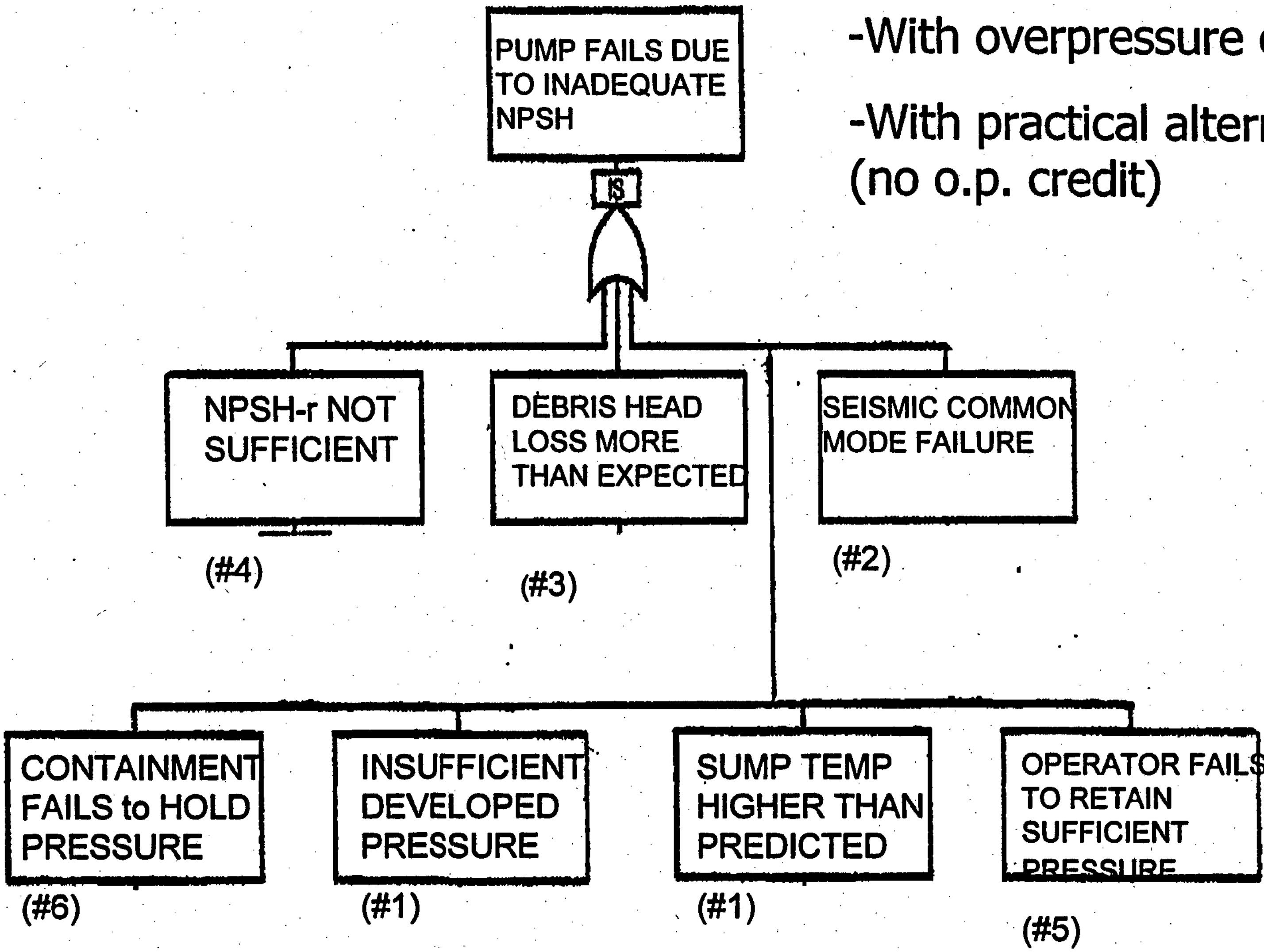
- On Oct 7, 2005, Dr. Brian Sheron proposed a risk based (RG 1.174) approach in lieu of implementing practical alternatives
- We believe this approach may have promise
- Entergy analyzed part of the problem, but not the whole problem. An analysis of the whole problem would shed light on the risk of the overpressure proposal

NEW TOP EVENT

Two Cases:

-With overpressure credit

-With practical alternative (no o.p. credit)



NRC FORM 366 (6-2004) | U.S. NUCLEAR REGULATORY COMMISSION

APPROVED BY OMB: NO. 3150-0104 | EXPIRES: 06/30/2007

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LICENSEE EVENT REPORT (LER)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000 271	1 OF 3

4. TITLE
Primary Containment Leak Rate Testing Program Second Barrier Isolation Valve Found Miss-Positioned

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	
10	04	2005	2005	- 002 -	00	12	01	2005	FACILITY NAME N/A	DOCKET NUMBER

9. OPERATING MODE
N

10. POWER LEVEL
100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☐ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☒ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

CONTACT NAME	TELEPHONE NUMBER (Include Area Code)
William F. Maguire, General Manager Plant Operations	(802) 257-7711

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
N/A					N/A				

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	☒ NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On 10/04/05, with the reactor at full power, a sample line isolation valve, V10-198A on the Residual Heat Removal (RHR) "A" Loop was found open. V10-198A is a second barrier that supports Primary Containment Integrity and is required to be closed per the Primary Containment Leakage Rate Testing Program (PCLRTP). Upon discovery, Operators closed V10-198A and placed it under administrative control. This condition was identified while reviewing a Safety Classification Worksheet for a different valve in the RHR system sample line. The RHR System procedure valve line-up listed V10-198A as "open" and the RHR System Piping and Instrumentation Diagram (P&ID) displayed it as "closed". The open valve provided a potential flow path of water from Primary Containment to Secondary Containment. Two air operated valves and a manual sample valve located downstream of V10-198A provided reasonable assurance that effective isolation for this flow path was maintained during plant operation. The cause of this condition was the application of an insufficient change process in 1996 during implementation of the Qualified Closed Loop Outside Primary Containment modification that lacked sufficient documentation and reviews to effectively implement the change. There was no significant increase in radiological risk to plant workers or the public as a result of this condition.

NRC FORM 366 (6-2004) | PRINTED ON RECYCLED PAPER

Example this week of a containment isolation valve left open for 10 years

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10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

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N/A					N/A				

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
○ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)*	◉ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On 10/04/05, with the reactor at full power, a sample line isolation valve, V10-198A on the Residual Heat Removal (RHR) "A" Loop was found open. V10-198A is a second barrier that supports Primary Containment integrity and is required to be closed per the Primary Containment Leakage Rate Testing Program (PCLRTP). Upon discovery, Operators closed V10-198A and placed it under administrative control. This condition was identified while reviewing a Safety Classification Worksheet for a different valve in the RHR system sample line. The RHR System procedure valve line-up listed V10-198A as "open" and the RHR System Piping and Instrumentation Diagram (P&ID) displayed it as "closed". The open valve provided a potential flow path of water from Primary Containment to Secondary Containment. Two air operated valves and a manual sample valve located downstream of V10-198A provided reasonable assurance that effective isolation for this flow path was maintained during plant operation. The cause of this condition was the application of an insufficient change process in 1996 during implementation of the Qualified Closed Loop Outside Primary Containment modification that lacked sufficient documentation and reviews to effectively implement the change. There was no significant increase in radiological risk to plant workers or the public as a result of this condition.

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VERMONT YANKEE NUCLEAR POWER STATION (VY)	05000 271	2005	- 002 -	00	2 OF 3

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

DESCRIPTION

On 10/04/05, with the reactor at full power, a 3/4" manual globe valve for the Residual Heat Removal (RHR) "A" Loop sample line (V10-198A) was found open. V10-198A is a second barrier to Primary Containment and is required to be closed as necessary to maintain the RHR system water seal during plant operation per the Primary Containment Leakage Rate Testing Program (PCLRTP). This condition was discovered while reviewing a Safety Classification Worksheet for a different valve in the RHR sample line. The RHR System procedure valve line-up listed V10-198A as "open" and the RHR System Piping and Instrumentation Diagram (P&ID) displayed the valve as "closed". The open valve provided a potential flow path of water from Primary Containment to Secondary Containment for the water seal that serves as part of the second barrier for Primary Containment, during a Design Basis Loss of Coolant Accident with a concurrent seismic event.

Upon discovery of this condition, Operators closed V10-198A and placed it under administrative control by tagging the valve "SHUT". Two normally closed air operated valves (AOV) and a normally closed manual sample valve located downstream of V10-198A provided reasonable assurance that effective isolation for this flow path was maintained during plant operation. The RHR "keep fill" line maintains system pressure during normal operation to continuously demonstrate that Primary Containment Integrity is maintained. Any leakage through the series of closed valves would have been into the Reactor Building Sample Sink which is within the envelope of Secondary Containment and would be detected by Operations or Chemistry personnel.

The three valves located downstream of V10-198A are not credited as Primary Containment Isolation valves within the Program Procedure for the PCLRTP. However, both of the in-line AOVs close on a Primary Containment Isolation System (PCIS) signal and are designed with a fail-safe feature to close on a loss of instrument air. The manual sample valve located downstream of the PCIS valves is also maintained in the closed position. Additionally, the first AOV in the series, FCV10-160, is designed to perform during and after a design bases seismic event.

This condition was determined to be reportable to the NRC as a Condition Prohibited by Technical Specifications in accordance with 10CFR50.73(a)(2)(i)(B). VY Technical Specification 3.7.A.2 states that Primary Containment integrity shall be maintained at all times when the reactor is critical. Technical Specification 4.7.A.2 provides a surveillance requirement to ensure that this is accomplished by stating that Primary Containment integrity shall be demonstrated by the PCLRTP. Also, Technical Specification Definition 1.0.N. for Primary Containment Integrity states that all manual containment isolation valves that are not required to be open during accident conditions, are closed, and may be opened intermittently under administrative controls. V10-198A was not in the required closed position prior to discovery of this condition and was not administratively controlled open by a dedicated operator.

CAUSE

The root cause of this condition was determined to be the application of an insufficient change process (Job Order File process) that was utilized in October of 1996 during implementation of the Qualified Closed Loop Outside Primary Containment modification. The process that was used lacked sufficient documentation and reviews to effectively implement the change.

Contributing Causes included the following;

1) The inter-relationships between P&ID valve position, operating procedure valve position, and locked valve criteria were not well understood when the event occurred.

2) Thirty five successive revisions up to 1989 to the subject P&ID reduced the sharpness of the image quality for V10-198A causing the valve's normal "open" position to appear as "closed".

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

ASSESSMENT OF SAFETY CONSEQUENCES

The subject valve is a second barrier for Primary Containment per the PCLRTP. The first barrier valves remained operable and closed as required. There are two AOVs located downstream of V10-198A that are designed to automatically close on a PCIS signal or on a loss of instrument air. A third manually operated sampling isolation valve located downstream of the two AOVs is maintained in the closed position. Additionally, incidental leakage from the system past these three valves would be detected by Operations or Chemistry personnel at the Reactor Building Sample Sink. Therefore, reasonable assurance existed that Primary Containment Integrity was maintained. This condition did not result in a significant increase in radiological risk or industrial risk to plant workers or the general public in the event of a design bases accident.

CORRECTIVE ACTIONS

The Job Order File process that was used when this condition occurred was superseded by an improved design control process. The procedures that implement the current design control process provide clearer and more concise direction that would likely have prevented this condition from occurring if utilized in 1996.

Immediate Actions

1) Upon discovery and confirmation of this condition, V10-198A was closed and administratively tagged "SHUT".

Interim Actions

1) V10-198A was added to the "Current System Valve and Breaker Line Up and Identification" procedure controlled population.
2) A drawing change was submitted for the subject P&ID to indicate V10-198A/B normal positions as locked closed.
3) The RHR System procedure's appendix for normal system line up was changed to control V10-198A as closed.
4) The RHR and Core Spray system procedures were verified to ensure that the valve line-ups contained within them are in agreement with the procedure for the PCLRTP. No additional discrepancies were noted.
5) On November 10, 2005, the Vice President of Engineering distributed a memo to all Vermont Yankee site employees titled "Configuration Control at Vermont Yankee". This correspondence described the event, expectations for configuration control, current design control processes employed within the Entergy Fleet, provided a list reference materials and described the relevant points from the reference materials that need to be reinforced to prevent this type of event from recurring.
6) Radiation Protection containment sampling procedures were reviewed to ensure compliance with TS 1.0.N.1 and the PCLRTP Procedure administrative controls for manual containment isolation valves. No discrepancies were noted.

Long Term Actions

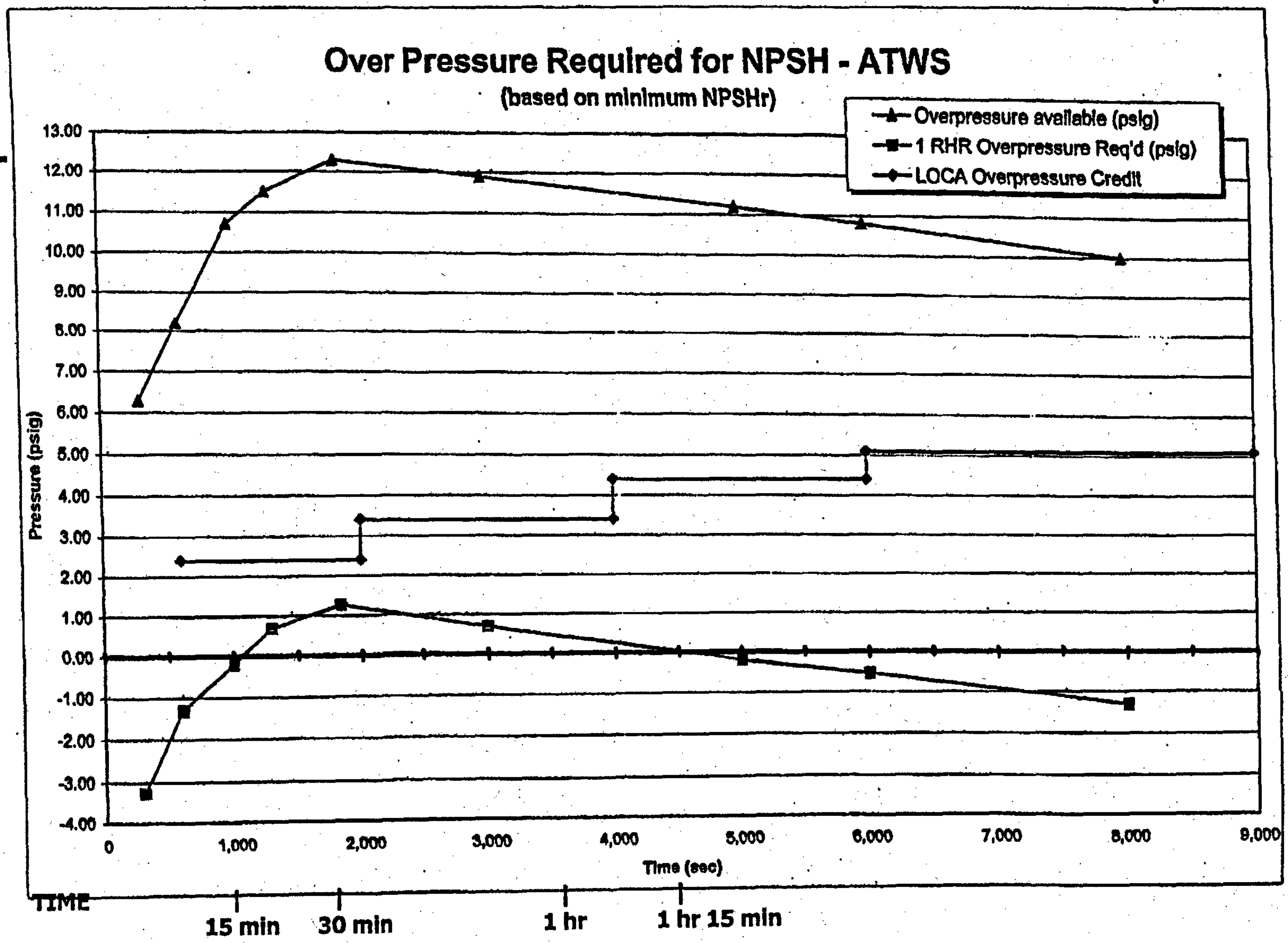
1) A review of other Job Order File changes from the same time frame will be performed to assess the potential for similar conditions.
2) Evaluate the need to review and as necessary correct the image quality and valve positions for the Control Room P&IDs referenced in the PCLRTP procedure.

ADDITIONAL INFORMATION

No similar events have occurred at Vermont Yankee within the past ten years.

ATWS NPSH Evaluation Deserves More Questions

- Asking overpressure credit for 1.25 hours
- More energy in ATWS than LOCA
- Pressure developed differently
- The type of conservative assumptions used for LOCA are not employed for ATWS


Over Pressure Required for NPSH - ATWS
(based on minimum NPSHr)
Overpressure available (psig)
1 RHR Overpressure Req'd (psig)
LOCA Overpressure Credit
13.00
12.00
11.00
10.00
9.00
8.00
7.00
6.00
5.00
4.00
3.00
2.00
1.00
0.00
-1.00
-2.00
-3.00
-4.00
Pressure (psig)
0
1,000
2,000
3,000
4,000
5,000
6,000
7,000
8,000
9,000
Time (sec)
TIME
15 min
30 min
1 hr
1 hr 15 min

SUMMARY

- Under the ACRS Letter of 9-20-05, overpressure credit should not be granted.
- Under Dr. Sheron's proposal:

1. Modification of the defense-in-depth concept is troublesome
2. Entergy analyzed part of the problem, but not the whole problem. An analysis of the whole problem would shed light on the risk of the overpressure proposal.

Pilot Engineering and Design Inspection at Vermont Yankee

Raymond Shadis

New England Coalition

BEFORE A MEETING OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

December 7, 2005

Rockville, Maryland

COMPARING NRC INSPECTION AND IEA

VPSB REQUESTED IEA

- deep vertical slice of 4 systems
- 2 safety systems
- 2 maintenance rule systems
- implies EOC review
- level of effort – 4 persons/4 weeks/40 hours or 640 dedicated hours
- independent

NRC DID PILOT ETI

- 45 components and operations of high risk/low margin
- eight issues were found
- limited extent of condition review
- 910 hours-500 routine + 410 pilot
- independent (2 yr)

COMPARING GOALS AND RESULTS

VPSB

- INDICATORS OF RELIABILITY
- ADDRESS PUBLIC CONCERN FOR SAFETY AND CALLS FOR AN ISA

NRC

- QUALITY OF ENGINEERING
- CONFIRM DESIGN AND OPERATION
- FIND AND FIX
- DID NOT PROBE AREAS WHERE PROBLEMS HAVE OCCURRED AT VY & OTHER EPU PLANTS

Staff Conclusions – Secy-05-0118

- NRC staff reported- pilot inspection to be improvement over ROP periodic inspection -found 8 issues per 1000 hours , old program found 4.5 per 1000 hours.
- Staff recommends engineering and design basis inspection be added to EPU review
- Staff found eight safety issues in 45 components/actions examined-small sample, large yield
- Staff stated that VY issues would not have been found in routine inspection

Comparing Pilot Inspection and Diagnostic Evaluation Team or ISA

PILOT INSPECTION

- small biased sample- high risk/low margin
- some focus on epu
- team members unfamiliar with epu and/or vy
- limited extentof condition review

DIAGNOSTIC EVAL.

- large sample by system type
- team members experienced in diagnostic evaluation
- large horizontal component- extent of condition/cause

Comparing Pilot Inspection and Diagnostic Evaluation Team or ISA...continued

PILOT INSPECTION

- Independence was marked by 2 year seperation from Vermont Yankee and Entergy

DET or ISA

- Maine Yankee ISA excluded NRC personnel from Region I and the Office of Nuclear Reactor Regulation